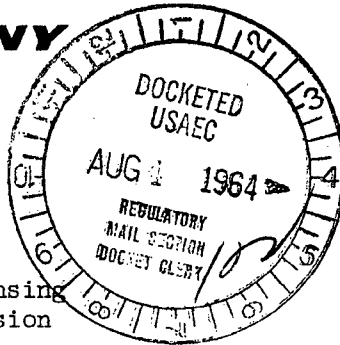


# MARTIN COMPANY

NUCLEAR  
DIVISION  
Baltimore,  
Maryland  
21203

In reply refer to:  
ACC-325



Mail No. 845  
August 4, 1964

Division of Materials Licensing  
U. S. Atomic Energy Commission  
Washington, D. C. 20545

DOCKET NO. 70-58

Attention: Mr. Donald A. Nussbaumer, Chief  
Source & Special Nuclear Materials Branch

*File Copy*

Subject: Proposed Amendment No. 21 to Special  
Nuclear Material License No. 53

Enclosure: (1) Proposed Amendment No. 21 to  
SNM-53 (six copies)

Gentlemen:

We are pleased to again submit a revised submission of the proposed Amendment No. 21 to Martin Marietta Special Nuclear Material License No. 53. The basic revisions from our July 9, 1964 submission are in our Nuclear Safety evaluations which result from discussions held with Messrs. Lauterbach and McCreless in our July 21, 1964 meeting at Bethesda. Included are other pages of our July 9, 1964 submission, with minor editorial changes, to present a complete picture of all aspects of our operation.

We appreciate the guidance given to us in Nuclear Safety aspects by Mr. McCreless and believe we have followed his suggestions in our re-evaluation. Approval of this proposed amendment will be needed by August 10 in order to enable us to start limited production shortly thereafter. We trust that our meeting on July 21 cleared up informative type questions, but please feel free to contact us if we can be of further assistance in this matter.

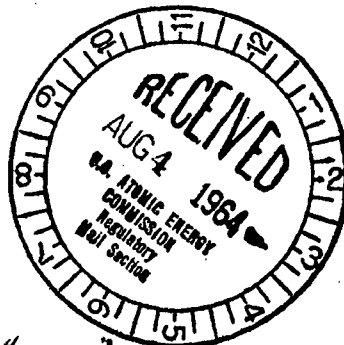
Very truly yours,

*C. W. Keller*

C. W. Keller  
Nuclear Accountability and  
Licensing Representative

CWK:lnl

Encl.



*B/27*

ACKNOWLEDGED

2 Copy Provided Compliance  
1 POR 108 REX 8/17/64

A DIVISION OF  
**MARTIN**  
**MARIETTA**

PROPOSED AMENDMENT NO. 21

TO

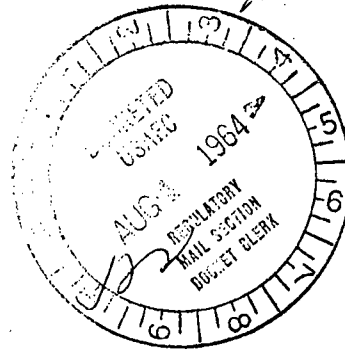
SPECIAL NUCLEAR MATERIAL *File Copy*

LICENSE NO. 53

DOCKET

*70-58*

*Trans. w/ 8-4-64*



ADMINISTRATIVE

4039

ANNEX 21  
TO AIRCRAFT WEIGHT AND BALANCE MANUAL NO. 23

It is the purpose of this proposed Amendment No. 21 to AIR-43 to describe the proposed control criteria, which will be used in the selection of a suitable type of tubular design employing slightly stretched metal wire.

DESIGN AND MATERIALS

1. The material shall be approximately 127 lbs. U-235.
2. Material - 303 stainless steel (AISI 303) and 304 (AISI 304).
3. The material shall be in the form of wire.
4. The material shall be in the form of wire.
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20. The material shall be in the form of wire (AISI 303 and AISI 304).

7. G. 1. 1. 1.
8. In presence of angle of steel tubes.
9. Accuracy of final element.
10. Accuracy of final element.

Method of Manufacture - General Description

Manufacture is defined as any operation which is performed which a specific fabrication step is performed. Each operation and unit shall be designated by a number or letter and by a unit number or two (2). The unit shall be well defined and shall be clearly and easily in which basic segments are subdivided into final elements shall be separated by a minimum distance of five feet. In other instances units are separated by a threshold corner, the shall be possible to achieve that unit area limit are not exceeded. Limiting unit spacing criteria are used to establish the U-235 limit in any unit area and for any operation.

The dimensions in this document will be given in feet and inches in most cases and include the pertinent number safety standards. Figure 1 shows the process flow and the U-235 limits assigned to each step.

Page 5 redacted for the following reason:

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(b)(4)

Statement of the Commission

The Commission has the honor to acknowledge the receipt of your letter of the 15th of June, 1954, in which you request that the Commission should consider the possibility of a study of the economic and social conditions of the people of the Republic of the Congo.

The Commission has the honor to inform you that it has decided to study the economic and social conditions of the people of the Republic of the Congo. The Commission has decided to study the economic and social conditions of the people of the Republic of the Congo. The Commission has decided to study the economic and social conditions of the people of the Republic of the Congo.

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Microscopic analysis of the specimen

The specimen was examined under a light microscope and the following observations were made:

The specimen was found to be composed of a mixture of small, irregular particles and larger, more uniform fragments. The smaller particles were dark in color, while the larger fragments were light brown. The overall appearance was that of a heterogeneous mixture.

The larger fragments were found to have a crystalline structure, as evidenced by their sharp edges and flat surfaces. The smaller particles were more amorphous in nature, with irregular shapes and sizes. The mixture was found to be stable under the conditions of the experiment.

Analysis

Further analysis of the specimen was conducted using X-ray diffraction and infrared spectroscopy. The X-ray diffraction pattern showed a series of sharp peaks, indicating a high degree of crystallinity. The infrared spectrum showed characteristic absorption bands for the material.

Conclusions

The results of the analysis indicate that the specimen is primarily composed of a crystalline material. The presence of amorphous particles suggests a complex or multi-phase structure.

The X-ray diffraction and infrared spectroscopy results are consistent with the proposed chemical structure of the material. Further studies are required to confirm the exact composition and properties of the specimen.

\* Reference - Proposed Amendment to 10 CFR, Part 70.

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NUCLEAR FUEL ELEMENTS  
FOR POWER REACTOR SYSTEMS

- A. Problem Fifty-two fuel elements will be arranged in a core through 12x12x36 inches long during the clearing operation of the fuel element fabrication process. The fuel element will be loaded on two racks, 26 fuel elements per rack, supported by a 12-inch fixed mechanical spacer. Part of the  $K_{eff}$  of this reflected system is shown in Figure 10. A homogeneous solution is assumed for some of the following equations; however, correction for low enriched heterogeneity is incorporated in the calculation of  $k_{eff}$ .
- B. Given Weight of U-235 per element = 10.968 gram.  
Weight of U-238 per element = 381.032 gram.
- C. Analytical  
Initial: The calculations were based on a unit cell 1.0 x 1.75 x 36 inches long. Twenty-six unit cells were by the active fuel volume analyzed. These unit cell dimensions were based on the spacing of fuel elements on the rack in the clearing stage. The rack is shown in Figure 10.

Material	Volume (in <sup>3</sup> )	Volume Fraction	Effective $k_{eff}$ (in <sup>-1</sup> )	Homogenized $k_{eff}$ (in <sup>-1</sup> )
H <sub>2</sub> O	23,795	.8846	.0211	.01955
U-235	57	.00212	2,470	.0548
U-238	1,225	.0456	.128	.00781
BY SS	586	.01938	.263	.00335
Oxygen & Impurities	1,228	.04577	0	0
Void	50	.00185	0	0

- D. For the age diffusion method of calculating steady state reactor conditions, the standard effective multiplication factor for a light water moderated system is given by:

$$k_{eff} = \frac{k_{inf}}{(1 + L_1^2 B^2) (1 + L_2^2 B^2) (1 + L^2 B^2)} \quad (\text{Ref. 2})$$

where  $k_{inf} = \epsilon p f \eta$

- and  $\epsilon$  = Fast fission factor  
 $p$  = Resonance escape  
 $f$  = Thermal utilization  
 $\eta$  = Neutrons liberated per thermal neutron absorbed in fuel.

E. The heat-transfer factor for slightly enriched uranium dioxide in water can be approximated by:

$$\xi = 1 + \frac{0.196}{1 + 0.60 \rho_w \left( \frac{V_w}{V_u} \right) + 0.208 \left( \frac{V_c}{V_u} \right)} \quad (\text{Ref. 5})$$

where  $\rho_w$  is the density of water in g/cc,  $V_w/V_u$  is the volume ratio of the water to uranium, and  $V_c/V_u$  is the volume ratio of the cladding to uranium.

$$\xi = 1 + \frac{0.196}{1 + (0.60)(1.0)(19.3) + 0.208(2.0)}$$

$$\xi = 1.012$$

The heat transfer factor as a function of ring radius for natural uranium is shown graphically in Reference 4. Using this data  $\xi$  is 1.015.

F. The resonance escape probability for a heterogeneous system can be approximated by:

$$p(E) \approx \exp \left( - \frac{f_{R_0}}{1 - f_{R_0}} \right) \quad (\text{Ref. 5})$$

$$\text{where } \frac{1}{f_{R_0}} = 1 + \frac{V_c}{V_u} \frac{\sum_{i=1}^N \Sigma_{a_i}}{\Sigma_{a_{u_0}}} F_{R_0} + (\bar{\epsilon}_{R_0} - 1), \quad \frac{V_c}{V_u} = 9.16 \quad (\text{Ref. 6})$$

$$\Sigma_{a_{u_0}} = \frac{\sum_{i=1}^N \Sigma_{a_i}}{\ln(\bar{\epsilon}_{R_0})} = \frac{1.58}{\ln(\bar{\epsilon}_{R_0})} \quad (\text{Ref. 7})$$

and values of  $\bar{\epsilon}_{R_0}$  were obtained from the Nuclear Engineering Handbook by Wilmington, pages 2-9, Table 13.

$$\Sigma_{a_{u_0}} = \frac{2b(a + b \sqrt{f_1})}{\ln(\bar{\epsilon}_{R_0})} = \frac{0.023(11.0 + 12.15 \sqrt{0.318})}{\ln(\bar{\epsilon}_{R_0})} = \frac{0.434}{\ln \bar{\epsilon}_{R_0}} \quad (\text{Ref. 8})$$

where  $a = 11.0$  and  $b = 21.5$  for uranium dioxide.

$$F_{R_0} = 1 + \frac{(K_0 A_0)^2}{8} = \frac{(K_0 A_0)^2}{192} \approx 1.004 \quad (\text{Ref. 9})$$

where  $\nu_0 = 0.31 \text{ cm}^{-1}$  for uranium dioxide

$$A_0 = 1.880$$

$$E_n = 1 + \frac{(k_1 \nu_0)^2}{2} \left[ \frac{\nu_0^2}{2\nu_1^2 - \nu_0^2} \ln \frac{\nu_0}{\nu_1} + \frac{1}{2} \left( \frac{\nu_0}{\nu_1} \right)^2 - \frac{3}{4} \right] \quad (2.9)$$

$$E_n = 1 + 1.887 \left[ \frac{1.071 \times 1.187}{3.235} + .009 \right] = 1.817$$

where  $\nu_1 = .885 \text{ cm}^{-1}$  for uranium dioxide

$$C_1 = 1.896$$

$$\text{and } \frac{1}{k_1} = 9.46 \left( -\frac{1.187}{3.235} \right) (1.896) = (1.817 - 1)$$

$$\frac{1}{k_1} = 12.022$$

$$\text{and } f_n = .0312$$

$$\text{Therefore } \rho(E) = \exp \left( -\frac{0.112}{3.235} \right) = .963$$

G. The thermal utilization for a homogeneous system can be represented by:

$$f_T = \frac{\sum_{i=1}^n \frac{f_{T,i}}{L_i}}{\sum_{i=1}^n \frac{f_{T,i}}{L_i} + \sum_{i=1}^n \frac{f_{T,i}}{L_{i0}} + \sum_{i=1}^n \frac{f_{T,i}}{L_{i\infty}}} = \frac{.0727}{.0727 + .3195 + 1.0135} = .721$$

H. The average number of neutrons,  $\eta$ , released for 4.5% enriched uranium is 1.928 (Ref. 10).

I. The reflector savings,  $\delta$ , for low-enriched, light-water moderated cores can be calculated from the following expression:

$$\delta \text{ (cm)} = 7.2 + 0.10 (W^2 - 40.0) = 7.11 \text{ cm} \quad (\text{Ref. 11})$$

$$\text{where } W^2 = L^2 + \tau = 39.12$$

and

$$\tau = 31$$

$$L = 2.85$$

the values of  $\tau$  and  $L$  were obtained from the Nuclear Engineering Handbook by Etherington, pages 2-9, table 13.

J. The buckling is calculated from the following expression:

$$B^2 = \left(\frac{\pi}{L}\right)^2 + \left(\frac{\pi}{W}\right)^2 + \left(\frac{\pi}{H}\right)^2 = .6329$$

$$\text{Width } a = 36 = 2.57 + 7.11 = 95.57 \text{ cm}$$

$$b = 7 = 1.0 \times 2.57 + 7.11 = 25.85 \text{ cm}$$

$$c = 7 = 1.75 \times 2.57 + 7.11 = 25.85 \text{ cm}$$

K. The thermal diffusion length is obtained from the following expression:

$$L^2 = L_0^2 (1 - f) = 2.019 \quad (\text{Ref. 12})$$

The thermal diffusion length,  $L_0$ , in water is equal to 2.85 cm. This quantity was obtained from the Nuclear Engineering Handbook by Etherington, pages 2-9, Table 13.

$$L_1^2 = 2.685$$

$$L_2^2 = 26.539$$

(Ref. 13)

The values of  $L_1^2$  and  $L_2^2$  are empirical constants derived from measurements on the slowing down of fast neutrons in water.

Therefore:

$$(1 + L_1^2 B^2)^2 = 1.1800$$

$$(1 + L_2^2 B^2) = 1.8726$$

$$(1 + L^2 B^2) = 1.0664$$

L. The calculated value for  $k_{eff}$  for the 26 fuel elements arranged as described in part C is:

$$k_{eff} = \frac{k_{\infty}}{(1 + L_1^2 B^2)^2 (1 + L_2^2 B^2) (1 + L^2 B^2)} = \frac{(1.012)(1.989)(751)(1.928)}{(1.1800)(1.8726)(1.0664)}$$

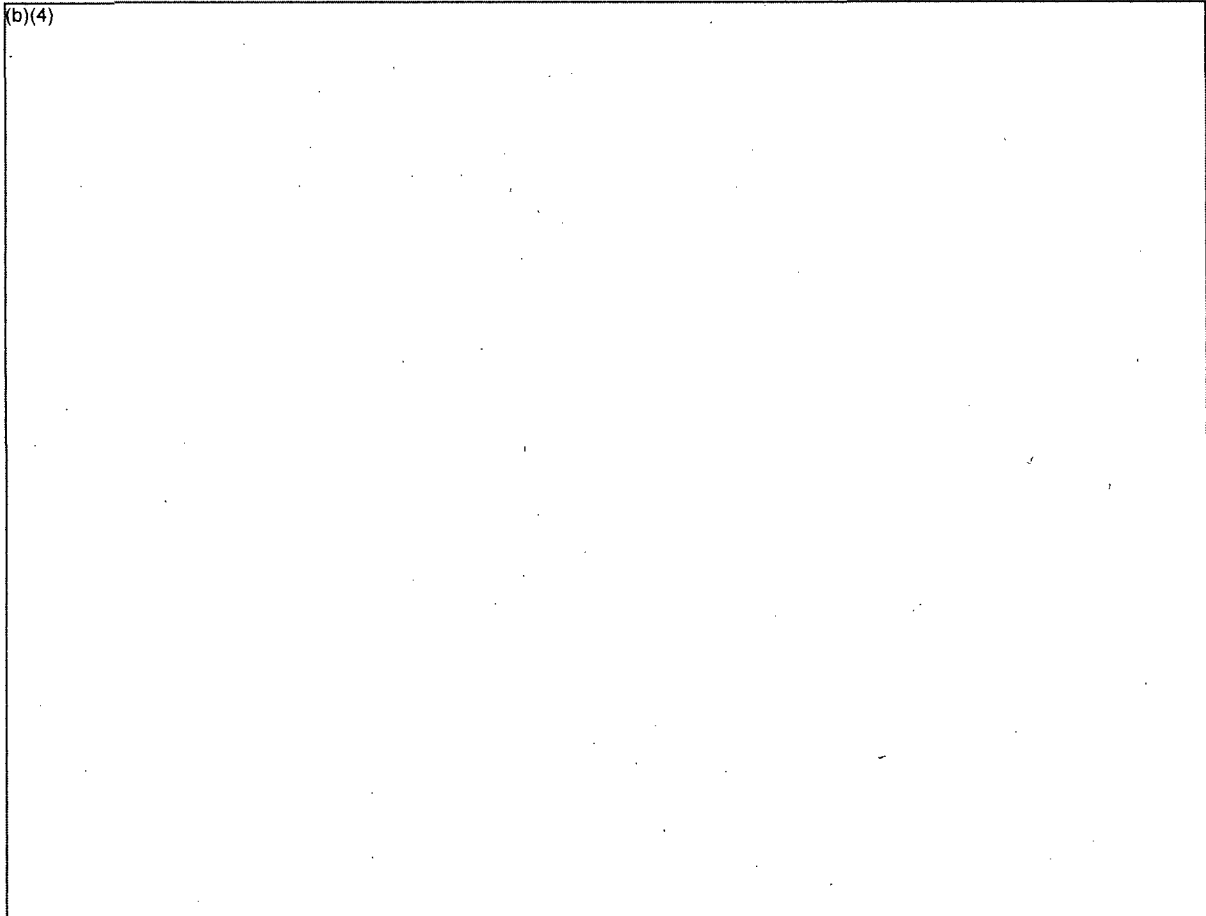
$k_{eff} = .602$  (this  $k_{eff}$  value ( $< 1$ ) is confirmed by HW-69273, Figure 13, which shows the minimum critical mass of 5% fuel and 0.5 inch diameter rods to be 3.75 lbs of U-235 under conditions of ideal spacing. Although the 26 tubes were known to be individually subcritical--as shown in HW-59273--the  $k_{eff}$  calculation was performed to determine the allowable solid angle.)

M. Based on this  $k_{eff}$ , the total interaction steradians ( $\Omega$ ) (Ref. 14)

allowed for an array calculation is  $(9-10 k_{eff})$  or 3.0.

32 7 1 6 3 5

(b)(4)



Ex4

When the 3 1/2 inch diameter shaft ends 2 3 3, and 4 are in  
and 4 1/2 inch diameter shaft 1. The proposed case of cases 2, 3, and  
4 1/2 inch diameter shaft 1. The distance from the center of  
shaft 1 to the edge of case 2 3 3 4 is 12, 30, and 35 inches  
the 3 1/2 inch diameter shaft angle is

3 1/2 inch diameter shaft 1 300 80 120  
144 300 1,044 2 70

3 1/2 inch diameter shaft 1 12 inch  
approximate distance is re-interpreted as total width  
of 24 inches and by mechanical use  
3 1/2 inch diameter shaft 1. This is a distance of 12  
inches from the center of shaft 1 to the edge of case 2 3 3 4

and Board of Health

When a lot of 10' fuel oil is used in connection with a building, they will be used in the same way as the fuel oil used in the building. It is understood that the building is to be used for the purpose of storing fuel oil and that the building is to be used for the purpose of storing fuel oil. The cost of the storage building is to be approximately \$1,000. A limit of one cubic foot per gallon will be permitted for the fuel oil stored in the building. The fuel oil will be stored in the building prior to the delivery of the fuel oil.

Capital fuel oil will be stored in the building which is to be used for the purpose of storing fuel oil. It is understood that the building is to be used for the purpose of storing fuel oil and that the building is to be used for the purpose of storing fuel oil. The cost of the storage building is to be approximately \$1,000. A limit of one cubic foot per gallon will be permitted for the fuel oil stored in the building. The fuel oil will be stored in the building prior to the delivery of the fuel oil.

References:

- (1) Melnikoff, Volume 20, No. 8 - August 1952, pp. 158-161.
- (2) Glasstone & Seaborn, Nuclear Reactor Engineering, p. 197, Equation 4.59.
- (3) Ibid, pp. 206-207.
- (4) Henny, Introduction to Nuclear Engineering, p. 122, Figure C.9.
- (5) Glasstone & Seaborn, Nuclear Reactor Engineering, p. 195, Equation 4.72.
- (6) Ibid, p. 197, Equation 4.63.
- (7) Ibid, p. 199, Equation 4.70.
- (8) Ibid, p. 196.
- (9) Ibid, p. 192.
- (10) EN-59473, p. 18, Appendix 1.
- (11) Deutch, Method For Analyzing Low-Enrichment, Light-Water Cores, Nuclear Science and Technology, Vol. 14, No. 8, pp. 150-173.
- (12) Glasstone & Seaborn, Nuclear Reactor Engineering, p. 183, Equation 4.56.
- (13) Ibid, p. 183.
- (14) Nuclear Safety Guide, EN-4016, Rev. 1 (1951), Figure 26.



Pages 17 through 27 redacted for the following reasons:

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(b)(4)