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TO D. G. Darr AT DATE May 28, 1970

FROM R. E. Kropp AT COPY TO

SUBJECT Criticality Safety Analysis of Shipping Container Model UNC-2800 with Four Dresden (BWR) Fuel Assemblies per Container

1.0 SUMMARY AND CONCLUSIONS

An analysis has been performed to determine the criticality safety of individual Shipping Containers Model UNC-2800. Then, arrays of these containers were evaluated. Each container will be loaded with four Dresden (BWR) fuel assemblies.

Individual packages are safe under normal transport conditions (dry) and have a $K_\infty = 0.77$ under accident conditions.

Arrays of packages also are safe under normal transport conditions (dry). Under accident conditions (optimum water between inner and outer container and between individual containers), an infinite array of packages has a $K_\infty = 0.818$.

Therefore, this container when loaded with four Dresden (BWR) fuel assemblies is subcritical under all possible conditions of moderation and reflection, and meets the criteria for a Fissile Class I Package.[1,2]

2.0 DESCRIPTION OF CONTAINERS

The Model UNC-2800 Shipping Container consists of a fuel assembly support nine inches high by 10 inches wide by 192 inches long, seven-gage steel "U" shaped strong-back with adjustable end clamps and cross support brackets. The fuel element support is shock mounted to an outer container by shear mounts. The outer container is a 12-gage steel cylinder 36 inches I.D. by 207 inches long with flange closure, skids, stacking brackets and roll rings. The container is described in Applied Design Company's Drawings, numbers 874A1-874A7, 874A99-874A114, and 874A116-874A136. The fuel assembly arrangement and clamping design will be as described in UNC Drawing D-304329, Rev. 1.

3.0 METHOD OF ANALYSIS

The Diffusion Coefficient and Macroscopic Cross Sections for the uranium/water/strong-back (stainless steel) region were calculated using the LOCALUX-3[3] one dimensional (cylindrical), multienergy unit cell or fuel assembly reactivity and depletion code. The LOCALUX [3,4,5,6] code is an improved version of the LASER[7] code. Fast and thermal energy spectra effective cross sections are obtained from an included MUFT[8]-THERMOS[9] calculation.

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The MUFT portion performs a homogeneous calculation in the energy range from 10 MEV to 1.855 ev. To account for lumping of absorbers and the resultant shielding of the U^{238} resonances, LOCALUX-3 provides for the application of self-shielding factors based on the Dancoff and Doppler effects. These self-shielding factors are calculated using the interaction method of Strawbridge and Barry.[10]

The THERMOS portion handles the spatial characteristics of the fuel rods discretely by integral transport theory so as to account for energy and spatial self-shielding. The thermal energy region covers the range from zero to 1.855 ev. The NELKIN[11] bound proton scattering kernel is used. A typical configuration treated by LOCALUX-3 is depicted in Figure 1.

All basic microscopic cross section data are contained within the LOCALUX libraries. These libraries were originally compiled by Westinghouse Atomic Power Division as part of the LASER code and have been revised by United Nuclear Corporation as reported experimental measurements become available. The LOCALUX codes or cross sections do not contain any arbitrary adjustments to force agreement with critical experiments.

The LOCALUX codes are used routinely at United Nuclear Corporation as a basic tool for generation of depletion dependent reactor constants for both BWR and PWR assemblies. The parent code, LASER, was checked extensively against criticals by Westinghouse Atomic Power Division. The results indicated that the LASER code was preferred over the LEOPARD[12] code from a standpoint of a smaller spread between calculations and experiments. The standard deviation in reactivity for mixed oxide (UO_2 + PuO_2) lattices was 6.19×10^{-3} for LASER and 9.26×10^{-3} for LEOPARD.[13] For UO_2 lattices, the standard deviation was 0.974×10^{-3} for LASER and 2.31×10^{-3} for LEOPARD.[13]

The LOCALUX codes have been checked extensively at United Nuclear against both PWR and BWR experimental data as well as against Monte Carlo calculations. A program has been completed at United Nuclear[14] which established the superiority of the LOCALUX (LASER) code over LEOPARD in calculating mixed oxide lattices. The program results were similar to those reported by Westinghouse.

Diffusion Coefficients and Macroscopic Cross Sections for the regions between the strong-back and outer container and outside the outer container were calculated using the FORM[15] and TEMPEST[16] codes. Conservative values of the Diffusion Coefficient were assumed for conditions of low water content (under-moderated) by clipping the rising values at 10 cm. Diffusion Coefficients and Macroscopic Cross Sections for the region within the outer container were calculated using LOCALUX-3.

Array infinite multiplication calculations were performed using AIM-6, a multigroup, one-dimensional diffusion theory code. AIM-6 is based primarily upon the AIM-5[17] code and differs from it in that provision is made for the use of a macroscopic cross section library.

Array calculations were performed assuming four regions: (1) the fuel-moderator-strong-back, (2) the space between the strong-back and the outer container, (3) the outer container and (4) the space outside the outer container. Two energy groups were used and zero boundary current was assumed.

4.0 NUCLEAR SAFETY EVALUATION - NORMAL TRANSPORT CONDITIONS

Under normal transport conditions, both the individual packages and arrays of packages are dry. There is no water or other moderating material within the fuel assembly, within the container voids, or outside the container. It has been shown that "...exponential experiments indicate that unmoderated uranium cannot become critical if the U-235 content is below 5 or 6 wt. %."[18] Therefore, both individual packages and arrays of packages are subcritical under normal conditions.

5.0 NUCLEAR SAFETY EVALUATION - ACCIDENT TRANSPORT CONDITIONS

Under accident conditions, it is assumed that the containers are damaged such that the outer mean diameter is reduced from 36 inches to 34 inches. The results of the structural tests indicate that the integrity of the strong-back and the fuel assemblies were resistant to the tests performed. Sufficient damage to the outer container seal was noted so that water in-leakage is possible.

The basis of the accident evaluation is:

1. Individual Containers
   a. Damaged container (O.D. = 34.0 inches)
   b. Full or optimum moderation of the fuel assemblies (H/U-235 = 165)
   c. Full density water between inner and outer container.
   d. Container reflected (4 cm of water)

2. Array
   a. Damaged container
   b. Full or optimum moderation of fuel assemblies (H/U-235 = 165)
   c. Variable water density between inner and outer container.
   d. Variable thicknesses of water outside outer container.

5.1 Individual Container

The maximum K_∞ = 0.770 as shown on Table I. Decreases in internal moderation and/or reduced reflection would result in decreased K_∞ values. Therefore, individual containers are subcritical.

5.2 Arrays of Containers

Infinite multiplication factors for arrays were calculated varying the density of water between the inner and outer containers and varying the thickness of water outside the outer container. The maximum K_∞ = 0.818 occurs with optimum internal moderation and no water between inner and outer container or between individual containers. Increasing water density between inner and outer container and increasing water thickness between individual containers results in decreased K_∞ values. These results are shown in Table I. Decreases in internal moderation would result in further decreases in K_∞ values. Therefore, arrays of containers are subcritical under all accident conditions.

6.0 CONCLUSIONS

The evaluation indicates that individual containers or arrays of containers are subcritical under all conditions. Thus, the container meets the requirement for a Fissile Class I Package.[1,2]

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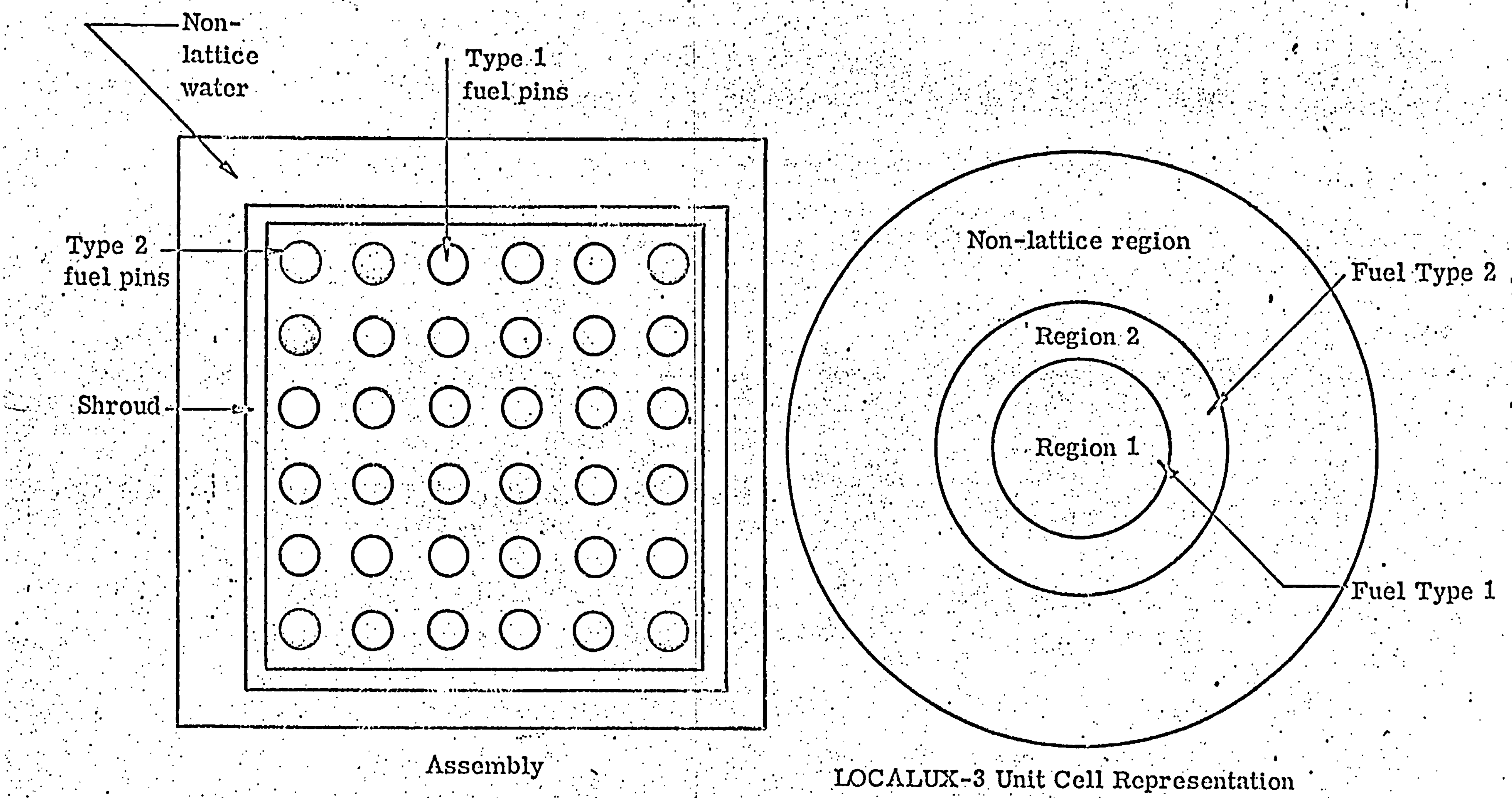

Fig. 1 — LOCALUX-3 Assembly Configurations

TABLE I

K_∞ VS AMOUNT OF WATER BETWEEN INNER AND OUTER CONTAINER AND WATER OUTSIDE THE OUTER CONTAINER

Amount of Water Between Inner and Outer Container (%)	Amount of Water Outside The Outer Container (cm)	K_∞
0.0	0.0	0.818
	1.0	0.744
	2.0	0.713
	3.0	0.695
	4.0	0.684
1.6	0.0	0.805
	1.0	0.741
	2.0	0.711
	3.0	0.694
3.22	0.0	0.788
	1.0	0.733
	2.0	0.708
	3.0	0.693
5.0	0.0	0.770
	1.0	0.726
	2.0	0.705
	3.0	0.692
10.0	0.0	0.736
	1.0	0.711
	2.0	0.696
	3.0	0.687
100.0	4.0	0.770*

* Flux assumed to be 0 at the boundary.

REFERENCES

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2. Title 49 Code of Federal Regulations, Part 173

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15. McGoff, D. J., "FORM - A Fourier Transform Fast Spectrum Code for the IBM-709", NAA-SR - Memo - 5766, September 1960

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Description of Dresden (BWR) Pellets, Rods and Elements

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

Description of Dresden (BWR) Pellets, Rods and Elements (continued)

2.

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

D. General Information

1. Enrichment

a. Normal Fuel 2.34 ± 0.05 w/o

b. Segmented Fuel 2.34 ± 0.05 w/o

c. PPC Fuel 1.77 ± 0.05 w/o

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

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

2. Pellet Density: 93.5% Theoretical ± 2.0%

Source of Data:

NIS-CP-LJS-22, dated October 31, 1968, "Renewal of Special Nuclear Materials License SNM-777, Docket 70-820. Section: 800, Subpart 823, Nuclear Safety Evaluation--Dresden-1 Fuel Elements (BWR), pagse 1 and 2 of 6.