5.2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet.

b. Annulus nominal volume = 543,000 cubic feet.

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 230.5 feet.

d. Nominal inside diameter = 148 feet.

e. Cylinder wall minimum thickness = 3 feet.

f. Dome minimum thickness = 2.5 feet.

g. Dome inside radius = 112 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment vessel is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

PENETRATIONS

5.2.3 Penetrations through the containment structure are designed and shall be maintained in accordance with the original design provisions contained in Sections 3.8.2.1.10 and 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircoloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 2250 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have maximum enrichment of 3.7 weight percent U-235.

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SAFETY EVALUATION

Re: St. Lucie Unit 1
Docket No. 50-335
Fuel Assembly Enrichment

Attachment A: Spent Fuel Storage Rack - Criticality Evaluation Summary

Attachment B: New Fuel Storage - Criticality Evaluation Summary

Attachment C: Fuel Inspection Elevator, Upender, & Fuel Transfer Tube - Criticality Evaluation Summary

10/4/79

Florida Power & Light
Spent Fuel Pool Storage Rack
Criticality Evaluation
Summary

I. PURPOSE & RESULTS

This report presents a summary of the criticality evaluation of the high capacity (HI-CAP™) fuel storage rack designed to accomodate 728 fuel assemblies in fuel storage locations in the spent fuel pool at the St. Lucie Nuclear Station, Unit 1. By virtue of the conservative assumptions employed in the criticality evaluation, it is concluded that under normal operating conditions and with a limiting UO_2 feed enrichment of 3.7 w/o U-235, the multiplication factor of the fully loaded rack in the flooded spent fuel pool does not exceed the limiting multiplication factor of 0.95 specified in ANSI-N210-1976. This conclusion is based on the results of analyses which predict a multiplication factor of 0.947 for the rack when fully loaded with fresh fuel of the limiting enrichment and immersed in pure water at 68°F, including allowances for calculational uncertainties and biases. These analyses employ arrays of storage cells that are of infinite extent in both the lateral and axial directions, and include the effects of the most adverse combination of mechanical tolerances and fuel displacement. Increasing the coolant temperature from 68 to 225°F decreases the multiplication factor by 0.015.

II. DISCUSSION

This report provides a description of the criticality evaluation of the high capacity fuel storage racks designed and manufactured by Combustion Engineering, Inc. for installation in the spent fuel pool of the St. Lucie Nuclear Station, Unit 1. This rack design provides 728 normal storage locations. Each storage location is designed to accommodate one fuel assembly consisting of 176 fuel rods, and 5 control rod guide tubes in a 14 x 14 array with a nominal pitch of 0.580 inches.

The fuel storage locations are of the HI-CAP™ design which, for this application, consists of a type 304 stainless steel box structure having a square cross sectional geometry with a nominal internal dimension of 8.4835 inches. The box walls which completely enclose the four vertical sides of a fuel assembly have a nominal thickness of 0.250 inches. The nominal center to center spacing of 12.53 inches for each fuel storage box, and the nominal water gap thickness of 3.548 inches between adjacent boxes, are maintained within specified tolerances by structural members welded to the exterior surfaces of the boxes.

Subsequent sections of this report discuss the design bases and results of the criticality evaluation.

III. DESIGN BASES

A. Multiplication Factor

The fuel storage rack is designed to meet the subcriticality margin specified in Section 5.1.12.1 of ANSI-N210-1976[1] which states in part -

5.1.12.1- "The spent fuel storage racks shall be designed to assure that a keff not greater than 0.95 is maintained with the racks fully loaded with fuel and flooded with unborated water. ---- The design shall be based on the maximum enrichment and fissile isotopic content of fuel to be cycled in the plant. ------".

B. Assumptions Employed in Criticality Evaluation

The following assumptions are employed in the criticality evaluation to assure that the evaluation is conservative over the range of fuel assembly design variables provided in the specification and/or anticipated operational conditions affecting the criticality margin of the spent fuel pool.

1. Neutron leakage effects are taken to be those characteristic of an infinite array of fully loaded, spent fuel storage locations in the lateral directions and infinitely long fuel assemblies and storage box walls in the axial direction. For the analyses of normal spent fuel locations employing the reference 8.4835 inch I.D. stainless steel box, an infinite array of storage cells having a nominal square dimension of 12.53 inches is employed. In these analyses it is assumed that each fuel storage location contains a fresh fuel assembly of the limiting enrichment (3.7 w/o U-235).

2. Parasitic neutron capture contributions in the fuel storage rack structural material are conservatively represented by neglecting all structural materials other than the stainless steel box walls.

3. The spent fuel pool is assumed to be flooded with pure (unborated) water at a temperature of 68°F. Elevated coolant temperature effects are assessed by evaluating the reactivity change between isothermal lattice calculations at 68 and 225°F. ....

4. Each fuel assembly is assumed to be loaded with unirradiated UO_2 having an enrichment of 3.7 w/o U-235. No burnable poison pins, control rods, or neutron sources are assumed to be present in the fuel assemblies.

5. Parasitic neutron capture contributions of structural components in the fuel assembly are conservatively represented by neglecting the zircaloy spacer sleeves and grids.

6. The effect of fuel storage rack mechanical tolerances and fuel assembly displacement within the fuel assembly storage box is calculated in a conservative fashion by assuming the most adverse concurrent combination of dimensional tolerances and a simultaneous diagonal displacement of the fuel assemblies in each cluster of four adjacent storage locations such that each fuel assembly is in contact with two side walls of each box and the spacing between each pair of the four fuel assemblies is minimized. The most adverse concurrent combination of dimensional tolerances corresponds to a configuration wherein the following conditions exist in each cell of the storage array: (1) mimimum pitch between centerlines of adjacent fuel storage boxes, (2) maximum storage box internal dimensions, and (3) minimum box wall thickness.

IV. HI-CAP RACK ANALYSES

A general description of the fuel storage rack in the spent fuel pool is given in Section III. The nominal dimensions of the normal fuel storage locations, defined by CE drawing [2] for the final reference design, are as follows:

I.D. of 304 stainless steel box, in.	8.4835
Thickness of steel box, in.	0.25
Water channel, in.	3.548
Center-to-center distance in.	12.5312

The physical parameters for the fuel assembly such as fuel pin radius and density, cell pitch, and composition of guide tubes are given in Table I.

The calculated multiplication factor for an infinite array of normal fuel storage locations, each containing one fuel assembly centered within the stainless steel box, is 0.8984.

To determine the most adverse effect of mechanical tolerances on the multiplication factor, the extremes in tolerances are used rather than a statistical model. The following tolerances and restraints apply to the nominal dimensions of the final reference design:

I.D. of steel box at top and bottom, in.	±0.0625
Minimum water channel, in.	2 55/64
Box wall thickness, in.	-0.01 and +0.047
Box wall bow, in.	± .250
Center-to-center spacing at top and bottom from corner of rack, in.	±0.125

To assess the effect of displacement of fuel assemblies within the storage boxes on the multiplication factor, each fuel assembly is assumed to be displaced diagonally against the corner of its storage box in a direction such that the closest distance of approach is achieved within each cluster of four storage boxes. Rack dimensions are assumed to be those corresponding to the minimum box wall case examined above. The calculated multiplication factor for this case is 0.9324.

To determine the reactivity at 150°F and 225°F for these analyses all materials and dimensions including the center-to-center spacing were expanded and thermal kernels at 150°F and 225°F were employed in the cross sections. An additional case at nominal dimensions at 68°F with the more normal 1720 ppm of dissolved boron present was also run.

The last two cases are used to determine the worth of the steel box for an isolated flooded assembly.

The following summarizes the results of the seven cases discussed above.

Case	Description	Box C-C Spacing	Box I.D.	Box Wall Thickness	Keff
1	Nominal Condition 68°F	12.4375*	8,4835	0.25"	0.8983
2	Minimum Offset Condition	12.0000**	8.6600	0.24"	0.9324
3	Nominal 150°F	12.4473	8.4902	0.25006"	0.8917
4	Nominal 225°F	12.4563	8.4963	0.25011"	0.8836
5	Nominal 68°F, 1720 ppm boron	12.4375	8.4835	0.25"	0.6751
6	Isolated, with steel box	32.98"	8.4835	0.25"	0.8075
7	Isolated, no steel box	32.98"	-	0.00"	0.8728

*The analyzed center-to-center nominal spacing is slightly smaller than the constructed value.
**Closest fuel assembly center-to-center spacing is 11.46".

The calculational uncertainties used in this evaluation consist of (1) a 0.0053 Δkeff uncertainty derived from comparisons of calculations for a series of UO_2 experiments, (2) a bias of 0.0019 in overpredicting criticality in these experiments, and (3) a bias in the calculating steel box wall worth inferred from calculations of the Johnson-Newlon experiments.[3] The magnitude of the latter bias is deduced in the following manner. The worth of the steel box wall structure which is obtained by subtracting the keff of case 6 (0.8075) from that of case 7 (0.8728) is found to be 0.0653 Δkeff. The analyses of the Johnson-Newlon experiment implied that for the addition of a 0.64 cm thick stainless steel shell to the uranyl fluoride solution container, the worth of steel was overestimated by a factor of 0.0041 divided by 0.0239 or 0.172. This factor times the calculated worth of steel box walls (0.0653) in the storage rack implies a calculational bias for the steel of 0.0112 Δk.

The reactivity balance for the criticality analysis of the normal spent fuel storage locations is summarized as follows:

Most adverse calculated Keff	0.9324
+95/95 confidence level calculational uncertainty	0.0053
+Bias UO_2 (Experiment - Calculation)	-0.0019
+Stainless Steel Calculational Bias	+0.0112
	0.9470

	Design Conditions	
	Nominal	Most Adverse
Multiplication Factor for Spent Fuel Storage Rack	0.8983	0.9470
Excess Margin	0.0371	0.0030

References

1. American Nuclear Society, Standards Committee Working Group ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations", ANSI-N210-1976, approved April 12, 1976.

2. CE Drawing #E-3077-667-002 Rev. 1, "Spent Fuel Rack Module", Rev. 1, April 11, 1977.

3. Clark, R. H., et al, Physics Verification Program Final Report B&W-3647-3 (March 1967).

TABLE 1

FUEL ASSEMBLY PARAMETERS

Fuel rod pitch, in.	0.58
Fuel rod array	14
Number of fuel rods per assembly	176
Fuel rod clad O.D., in.	0.440
Fuel rod clad I.D., in.	0.384
Fuel rod clad material	Zircaloy-4
Fuel pellet diameter, inc.	0.3765
Stacked fuel density, gm/cc	10.054
Number of control rod guide tubes per assembly	5
Guide tube material	Zircaloy-4
Guide tube O.D., in.	1.115
Guide tube I.D., in.	1.035
Fuel Assembly	
Active Fuel Height (in.)	136.7

ATTACHMENT B

Florida Power and Light New Fuel Storage - Criticality Evaluation Summary

PURPOSE & RESULTS

The purpose of this document is to present the results of a criticality evaluation made in 1974 in support of using the St. Lucie-1 new fuel storage rack for fresh UO_2 fuels with enrichments up to 3.7 w/o U235.

The new fuel storage racks consist of two arrays of 10 x 4 spaces for fuel assemblies separated by a 42-inch wide space as shown in Figure 1. The maximum effective neutron multiplication factor under conditions of uniform water (of any density <1 gm/cc) moderation in and between the assemblies should meet the requirements of Section 5.7.4.1 of ANS N18.2 which states:

> "The design of spent fuel storage racks and transfer equipment shall be such that the effective multiplication factor will not exceed 0.95 with new fuel of the highest anticipated enrichment in place assuming flooding with pure water. The design of normally dry new fuel storage racks shall be such that the effective multiplication factor will not exceed 0.98 with fuel of the highest anticipated enrichment in place assuming optimum moderation (e.g., a uniform density aqueous foam envelopment as the result of fire fighting). Credit may be taken for the inherent neutron absorbing effect of materials of construction or, if the requirements of Criterion 5.7.5.10 are met, for added nuclear poisons."

Typically, this type of array has a reactivity peak for full density water and a secondary peak in the water density range of 0.03 to 0.2 gm/cc. The rack, although normally dry with a keff of about 0.70, can be immersed in various water densities through fire fighting foams, floods, etc. For full density water the keff is 0.92, which is well within the requirements of a maximum keff of 0.95 including Physics uncertainties.

This study uses four-group transport calculations for water densities ranging between 0.02 and 0.075 gm/cc for the new fuel rack, indicating a maximum keff of about 0.89.

The physics uncertainties are much larger for these low density water systems than for flooded systems, since no applicable experiments have been performed. The calculated maximum keff's of about 0.89 would allow for an uncertainty of 9% to meet the 0.98 requirements of ANS N18.2. This is considered to be adequate.

Conditions Required for Criticality Safety

The calculations performed indicate that the proposed dry storage arrangements meet the criticality safety requirements of ANS Standard N18.2, with a margin of about 9% in keff for low density water moderation conditions. The proposed storage arrangements are, therefore, considered to be safe, subject to the following conditions:

1. Approved storage racks are used.
2. Criticality safety with plutonium recycle fuel has not been established.
3. The enrichment of the UO_2 assemblies is limited to 3.70 w/o U235.
4. The minimum surface-to-surface spacings between assemblies implicit in the analyses, are enforced in the rack specifications (see following section).

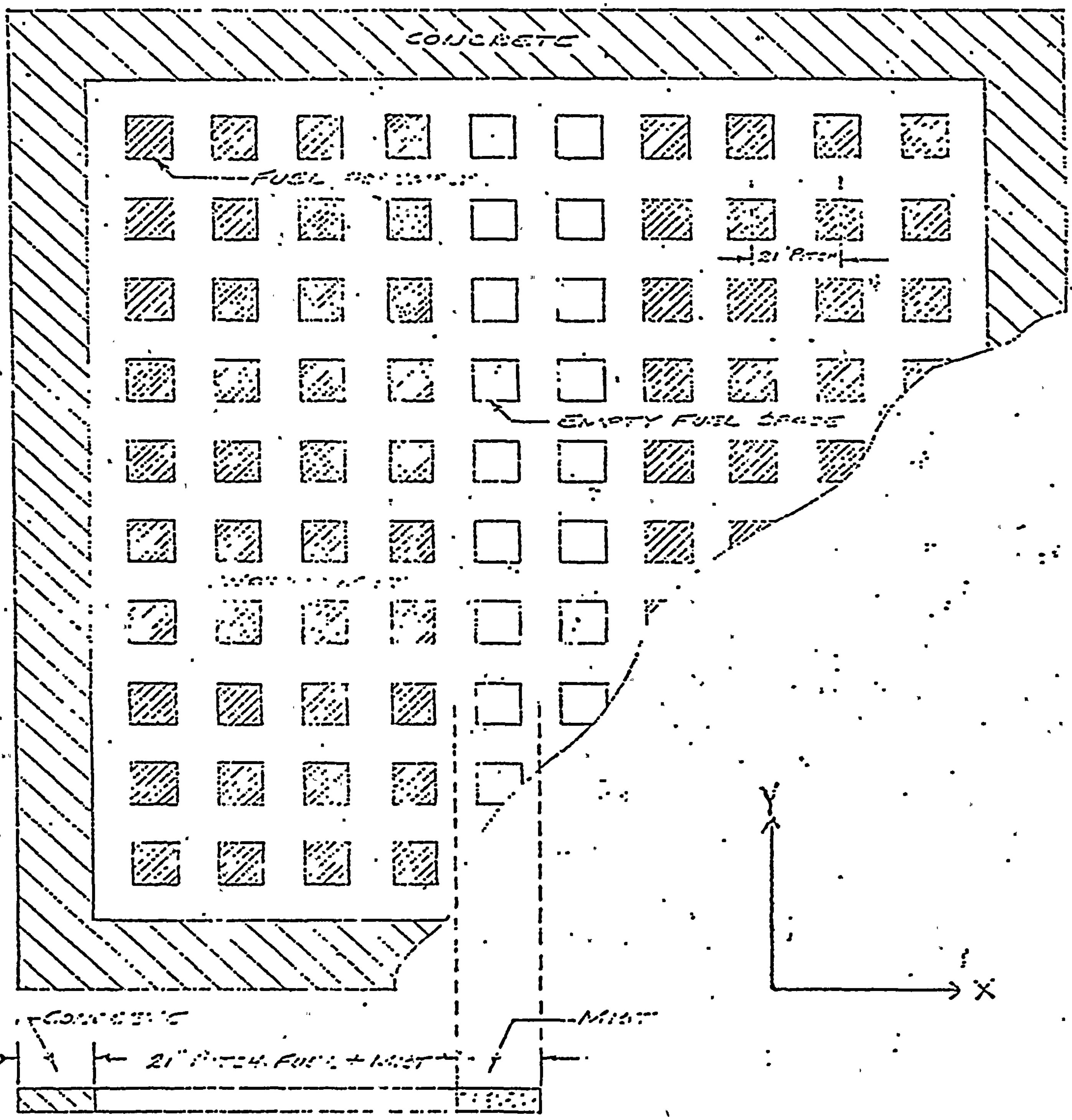
DISCUSSION

In its evaluation, CE has adopted the approach of assessing safety based on the minimum edge-to-edge spacing between any two assemblies, taking into account all dimensional tolerances and anticipated deformations during earthquakes, etc. On this basis, the minimum edge-to-edge spacing between assemblies would be greater than

21.00 - 0.50 (tolerance on pitch) - (8 15/16 + 1/16) = 11.5 inches,

rather than the 21.00 - 8.2 = 12.8 inches implicit in the analyses. It is estimated, that the reduction in spacing of 1.3 inches due to tolerances, plus an allowance for deformation (total spacing reduction estimated at 1.5 inches) everywhere would increase the maximum keff to about 0.916 , which would decrease the margin for Physics uncertainties to about 6% . However, the minimum spacing in one position would usually imply a larger spacing elsewhere, and thus the keff could increase to only .901 . The rack design is, therefore, judged to provide adequate criticality safety margins for conditions of fog moderation.

FIG. 1
REPRESENTATION OF
MODIFIED NEW FUEL STORAGE RACK
FOR FP&L ST LUCIE-1



ONE DIMENSIONAL ANISN REPRESENTATION

10/4/79

ATTACHMENT C

Florida Power & Light
Fuel Inspection Elevator,
Upender, Fuel Transfer Tube
Criticality Evaluation Summary

Purpose & Results

The purpose of this document is to provide a basis for updating Tech Spec 5.3.1 for St. Lucie 1 from an enrichment of 3.1 w/o to 3.7 w/o by presenting results of criticality analyses for the fuel inspection elevator, the upender, and the fuel transfer tube.

The applicable standard ANSI-N18.2 (Reference 1) section 5.7.41 states in part

> "The design of spent-fuel storage racks and transfer equipment shall be such that the effective multiplication factor will not exceed 0.95 with new fuel of the highest anticipated enrichment in place assuming flooding with pure water,"

The highest reactivity situation, assuming at least a four inch standoff to limit the approach of a second assembly, is 0.911, thus allowing a margin of more than 0.03 Δk beyond the allowance for calculational uncertainties determined by the analysis of a wide variety of critical experiments.

Design Input

The fuel dimensions and densities for the 14 x 14 pin assembly are taken from the St. Lucie 1 FSAR using a 3.7 w/o U-235 enrichment.

The fuel elevator dimensions are based on Programmed and Remote Systems Corp. Drawing # D-15699-D, Rev. B dated 6/29/78 of the elevator carriage. Standoffs appear in the Ebasco drawing 8770-6841 Rev. 1 to at least partially prevent a second assembly from approaching closer than a ten inch edge to edge separation. The steel structure was ignored in this analyses.

The upender dimensions are based on P.R.S.C. Drawing # A-13594-D Rev. D of 1971 for the Fuel Carrier Assembly and indicate that the closest approach, if two assemblies are in the carrier assembly, is 4 13/16 inches and also the presence of four 2x2x1/8 inch stainless steel full length angles at the corner of each assembly.

The fuel transfer tube inner radius of 35.25 inches was obtained from P.R.S.C. Drawing # A1-13499-D, Rev. E dated 7-28-76 of the Fuel Transfer Tube Rail Assembly Installation.

In all cases non-borated water at room temperature was assumed although normally a few thousand PPM of dissolved boron are present.

Discussion and Results

In order to more accurately predict the multiplication factor of the assembly arrays, reliable calculations of the spatial flux distribution, especially in the neutron absorbing steel regions, are essential. For this reason, a two dimensional transport calculation model of the transfer system is employed in which each component of the fuel transfer system geometry is explicitly represented. Thus, in the fuel upender calculation, the fuel assemblies, the water channel between the fuel assemblies, the steel angles, and the water reflector are represented as separate regions. The fuel assembly itself is represented as a 14x14 array of fuel assembly cells containing moderator and either fuel pins or guide tubes. Four neutron group cross sections are generated for each fuel assembly cell and for each component of the system with special attention given to the effect of adjoining regions on the spatial thermal spectrum and hence broad group thermal cross sections of each separate region.

The most reactive situation of the three considered would be for the fuel elevator when a second fuel assembly is assumed to be aligned with the one in the elevator with an edge to edge spacing of four inches, the resulting keff is 0.911.

For the upender the most reactive situation would be when a third assembly approaches to within four inches (edge to edge) of the two assemblies in the upender; the keff for this situation is 0.899. This keff is less than for the two assemblies separated by same distance in the elevator because the steel angles at each corner of both assemblies in the upender were included in the analyses.

The reactivity of the fuel array in the transfer tube will be less than for the case of the upender, i.e. a keff of <0.899. The reason being only two fuel assemblies can be in the transfer tube and the fuel is maintained in the same configuration as in the upender; a third assembly cannot approach the two assemblies while in the transfer tube.

The above multiplication factors are valid envelop values for a minimum separation between a third assembly for the upender and a second assembly for the fuel elevator of four inches or greater.

Reference:

1. American National Standards Institute "Nuclear Safety Criteria for the Design of Station and Pressurized Water Reactor Plants," ANSI-N18.2-1973, August 6, 1973.