ATTACHMENT 1

PROPOSED ZION APPENDIX A

TECHNICAL SPECIFICATON CHANGES TO

SECTION 5.0

DESIGN FEATURES

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5.0 DESIGN FEATURES

5.1 Site

Zion Units 1 and 2 are located at the Zion Station which consists of a tract of land of approximately 250 acres located in the extreme eastern portion of the city of Zion, Lake County, Illinois, on the west shore of Lake Michigan approximately 6 miles NNE of the center of the city of Waukegan, Illinois, and 8 miles south of the center of the city of Kenosha, Wisconsin. It is located at longitude 87° 48.1'W and latitude 42° 26.8'N.

The minimum distance from the reactor center line to the boundary of the exclusion area is 400 meters as shown in Figure 1.2-1 of the Zion FSAR and the low population zone is one mile as shown in figure 2.2-2 of the Zion FSAR, as defined in 10 CFR 100.3.

Reference

FSAR, Section 2

5.2 Reactor Coolant System

A reactor coolant system consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump, two isolation valves and a steam generator. The system also includes a pressurizer, connecting piping, pressurizer safety and relief valves, and a relief tank.

The Reactor Coolant System transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator.

The Reactor Coolant System provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values of uncontrolled release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor behavior. During transient operation, the system's heat capacity attenuates thermal transients generated by the core or steam generator.

The reactor coolant system components are designed in accordance with standards set in chapter 5 of the FSAR.

5.3 Reactor Core

The reactor is a multiregion core consisting of 193 fuel assemblies with 204 individual fuel rods per assembly.

A total of about .2 million lbs. of uranium dioxide with enrichments between 2.2 and 3.7 weight percent U-235 is contained in cold worked partially annealed Zircalloy tubes.

The equivalent core diameter is 132.7 inches and the active core height is 144 inches and generates 3250 MW_{th}.

The reactor containment structure for Zion Unit 2 essentially identical in design and construction to that of Unit 1 except that it is reoriented. Numerous mechanical and electrical systems penetrate the containment wall through welded steel penetrations.[2]

5.4.3 Containment Penetrations

All containment penetrations (both electrical and piping) are double barrier assemblies consisting of a closed sleeve, in most cases, or a double gasketed closure for special penetrations such as the fuel transfer tube. The space between the double barriers will be continuously pressurized, either by the Penetration Pressurization System or Nitrogen System, to a pressure in excess of the containment design pressure. [3]

References

(1) FSAR Section 5.1.1
(2) FSAR Section 5.1.2
(3) FSAR Section 5.1.4

5.5 Fuel Storage

5.5.1 New Fuel Storage

New fuel assemblies are stored in a separate storage vault which is designed to hold 132 new assemblies. The new fuel storage racks accommodate 2/3 of a core.

There are three sections of racks with each section made up of two rows. The two parallel rows in each section have a nominal center to center spacing of 21 inches and each section is separated by a distance of 44" to assure a K effective of less than 0.95 even for a condition of optimum moderation if water was to fill the vault, for fuel having a maximum loading of 50.2 gm U-235 per axial centimeter of fuel assembly length (about 4.0 weight percent U-235). The new fuel storage vault is protected from flooding by its free flood drain.

New fuel may also be temporarily stored in the spent fuel pool in preparation for refueling. The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 10.35 inches in both directions. This spacing is sufficient to maintain a K effective of less than .95 when flooded with unborated water, for fuel having a maximum loading of 46.4 gms. U-235 per axial centimeter of fuel assembly length (about 3.7 weight percent U-235).

5.5.2 Spent Fuel Storage

Irradiated fuel assemblies will be stored prior to offsite shipment in the stainless steel lined fuel pool which is located in the fuel handling building. Borated water is used to fill the spent fuel storage pit at a concentration to match that used in the reactor cavity and refueling canal during refueling operations. The fuel is stored in a vertical array with a nominal center to center spacing of 10.35" between assemblies to assure a K effective of less than 0.95 even if unborated water is used to fill the pit, for fuel having a maximum loading of 46.4 gms. U-235.

5.5.2 Spent Fuel Storage (Continued)

per axial centimeter of fuel assembly length (about 3.7 weight percent U-235).

Reference

Criticality Analysis of Zion Units 1 and 2, Fresh and Spent Fuel Racks, dated October 25, 1984.

5.6 Seismic Design

The structures, mechanical components and Engineered Safeguards Systems vital to safe shutdown and containment isolation, or whose failure might cause or increase the severity of a loss of coolant accident, are designed per the seismic criteria of Design Basis Earthquake (DBE). Design Basis Earthquake is based on ordinary allowable stresses as set forth in applicable codes, plus the additional requirement that a safe shutdown be made during a horizontal ground acceleration of 0.17g and a vertical acceleration of 0.11g occurring simultaneously. These systems and equipment are defined as Seismic Class 1.

Other systems and mechanical components in a support or auxiliary function are designed per the seismic criteria of Operational Basis Earthquake (OBE), or per applicable codes. These systems and equipment are defined as either Classes 2 or 3 depending on their function.

ATTACHMENT 2

The Westinghouse "Criticality Analysis of Zion Units 1 and 2 Fresh and Spent Fuel Racks" (Attachment 4) was prepared to support the request for licensing amendments to the Zion Units 1 and 2 facility operating licenses. The report addresses the neutron multiplication factor considering normal storage and handling of spent and fresh as well as postulated accidents with respect to criticality. The report follows the guidance of the NRC position paper entitled, "OT Positions For Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978, as amended by the NRC letter dated January 18, 1979. The thermal-hydraulic, mechanical, material, structural, and environmental aspects of the fresh and spent fuel racks, their layout and the mechanical fuel types being stored all remain unchanged.

In the criticality analysis of Zion spent fuel storage racks, Westinghouse assumed:

1) 15x15 OFA fuel, enriched to 3.7 w/o U-235, stored at its most reactive point of life;

2) pure water (no dissolved boron) at a density of 1 gram/cm^3;

3) an infinite array, both laterally and axially;

4) a minimum rack poison loading (.02 gm B-10/cm^2); and

5) a bias accounting for the B_4C particle self-shielding.

Mechanical uncertainties/tolerances arising from construction were treated using "worst case" conditions. No credit for fuel depletion, fission product buildup, presence of U-234 and U-236, or assembly grids was taken.

In the criticality analysis of Zion fresh fuel storage racks, Westinghouse assumed:

1) 15x15 OFA fuel, enriched to 4.0 w/o U-235, stored at its most reactive point in life;

2) pure water (no dissolved boron) at a density of 1 gram/cm^3 was assumed for normal conditions and .06 grams/cm^3 for "optimum moderation";

3) an infinite array model both laterally and axially was used for normal conditions and a finite array model was employed for "optimum moderation" conditions; and

4) credit was taken for neutron absorption in the stainless steel structural material for normal conditions but not for "optimum moderation" conditions.

Neither the spent fuel nor the fresh fuel rack analysis took credit for control rods or burnable poisons in the fuel. Worst case conditions for mechanical uncertainties and biases due to mechanical tolerances were also used. Refueling water boron concentration (2000 ppm) was only credited for accident conditions.

Considerations were given to various accident conditions to ascertain if they would increase the k_{eff} of the racks. It was determined by Westinghouse that most accident conditions would not result in an increased k_{eff}. For those accidents which could increase the spent fuel storage rack k_{eff}, the presence of 2000 ppm (minimum) soluble boron, which can be assumed as a realistic initial condition, precludes any postulated accident from exceeding a k_{eff} of 0.98. Accidents can also be postulated which would increase the k_{eff} of the New Fuel Vault. However, the most limiting single event is the introduction of moderation into the dry storage vault. Finite modeling of New Fuel Vault was performed under optimum moderation conditions and the results showed that the rack k_{eff} would be less than 0.98. For other accidents, the absence of water in the storage vault can be assumed as a realistic initial condition since assuming its presence would be a second unlikely event.

Attachment 4 demonstrates that the maximum effective multiplication factor (k_{eff}) for the Spent Fuel Pool during normal conditions is 0.9481. The maximum k_{eff} for the New Fuel Vault under normal conditions is 0.9086. As discussed above, the k_{eff} of both the Spent Fuel Pool and the New Fuel Vault during accident conditions is less than 0.98.

Since the results of Attachment 4 demonstrate that the k_{eff} of the normal storage of fuel will be less than 0.95 and that the k_{eff} during accident conditions will be less than 0.98, the storage of 3.7 and 4.0 w/o fuel is acceptable.

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ATTACHMENT 3

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

PROPOSED CHANGES TO ZION TECHNCIAL SPECIFICATIONS APPENDIX A - SECTION 5.0 DESIGN FEATURES

Commonwealth Edison Company has performed an evaluation of the hazards considerations associated with the proposed Technical Specifications amendment (Attachment 1). This evaluation has concluded that this change involves no significant hazards consideration. 10 CFR 50.92(c) states that a proposed amendment involves no significant hazards consideration if operation would not;

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) involve a significant reduction in a margin of safety.

The proposed amendment does not meet any of the significant hazards consideration standards of 10 CFR 50.92(c) and, therefore, a no significant hazards consideration findings by the NRC Staff is justified. The proposed amendment meets and exceeds example (vi) of the Commission guidance provided by 48 FR 14870.

> (vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

As discussed in Attachments 2 and 4, the results of the proposed change are clearly within all acceptable criteria. Specifically, the reactivity acceptance criteria of the Standard Review Plan, Sections 9.1.1 and 9.1.2, have been satisfied.

Therefore, since the application for amendment involves proposed changes that are similar to examples for which no significant hazards consideration exists, Commonwealth Edison Company has made a determination that the application involves no significant hazards consideration.

ATTACHMENT 4

CRITICALITY ANALYSIS OF ZION UNITS 1 and 2

FRESH AND SPENT FUEL RACKS

0294K

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Services Integration Division

Box 2728
Pittsburgh Pennsylvania 15230 2728

CWE-85-600

July 9, 1985

Mr. Everett Young
Commonwealth Edison Company
72 West Adams Street
Room 922
Chicago, Illinois 60690

Commonwealth Edison Company
Zion Units 1 and 2
Criticality Analysis of Zion Fuel Racks

Dear Mr. Young:

In response to your request, Westinghouse is pleased to release to Commonwealth Edison Company our recent report entitled "Criticality Analysis of Zion Units 1 and 2 and Spent Fuel Racks", as a non-proprietary document.

Should you have any questions, please do not hesitate to call me.

Very truly yours,

T H Sembach

for D. G. Maire, Manager
Central Area
Projects Department

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~~WESTINGHOUSE PROPRIETARY CLASS 2~~ P.L.

CRITICALITY ANALYSIS OF ZION UNITS 1&2
FRESH AND SPENT FUEL RACKS

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CE Core Design B

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1.0 INTRODUCTION

1.1 PURPOSE AND SUMMARY OF REPORT

The Commonwealth Edison Company is currently seeking a license amendment to raise the fuel enrichment storage limits of the spent fuel and fresh fuel storage racks used by the Zion Units 1 and 2. The mechanical construction and layout of the fresh and spent fuel storage area which is currently licensed will remain unchanged. This safety analysis report (SAR) has been prepared to support the request for licensing amendments to the Zion Units 1 and 2 facility operating licenses for use of the existing fresh and spent fuel racks that meet the criteria contained herein.

This report contains the revised nuclear criteria to which the existing racks are designed. It follows the guidance of the NRC position paper entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978, as amended by the NRC letter dated January 18, 1979.

The report addresses the neutron multiplication factor considering normal storage and handling of spent and fresh fuel as well as postulated accidents with respect to criticality.

The thermal-hydraulic, mechanical, material, structural and environmental aspects of the fresh and spent fuel racks are not considered in this report since the current fuel racks, their layout and fuel type being stored in the racks will remain unchanged.

1.2 GENERAL DESCRIPTION OF DESIGN FUEL RACKS

1.2.1 SPENT FUEL RACKS

The Zion Generating Station of the Commonwealth Edison Company consists of two pressurized water reactor generating units. The spent fuel storage pool is

shared by the two generating units. The present capacity of the storage pool is 2,112 spent fuel storage spaces (Figure 1-1).

The rack is made up of an array of tubes containing a neutron absorber material and are welded together along the length of the tubes with structural members or plates which provide the inter-tube connection. The center-to-center distance between assemblies is 10.35 inches.

The tube is stainless steel bearing boral, a neutron absorbing material produced by Brooks and Perkins. The tube is so constructed as to completely encapsulate and seal the absorber material. Each storage tube is 9.012 inches square in size (See Figure 1-2 for details).

The tube array forms a rigid structure and is mounted vertically and welded to a base plate. The base plate is supported by four legs and is elevated approximately 6 inches above the pool floor. Cooling water will flow through the hole provided in the plate and to the storage tubes to cool the storage cell (Figure 1-3).

1.2.2 FRESH FUEL RACKS

New fuel assemblies are stored in a dry vault which is designed to hold 132 assemblies in a fully loaded configuration. Within the vault there are three identical parallel sections of racks. Each section is capable of holding 44 assemblies and is made up of two different types of racks.

One type holds assemblies in a 6x2 array and the other holds assemblies in a 4x2 array. Within each section there are three racks of the 6x2 type and one rack of the 4x2 type. The four racks that make up each section are aligned such that they resemble a 22x2 rack. The center-to-center spacing of all fuel storage locations within a rack section is 21 inches in a fully loaded configuration. Each section is separated by a distance of 59 inches. The sides of the cells are open to the air with the exception of a few structural members for support.

1.3 CONCLUSIONS

On the basis of the design requirements presented in this report, operating experience with high density fuel storage and material referenced, it is concluded that the proposed increase of the Zion Units 1 and 2 fresh and spent fuel storage facilities enrichment limit will continue to provide safe fuel storage and that the modification is consistent with the facility design and operating criteria.

2.0 CRITICALITY ANALYSIS FOR ZION SPENT FUEL RACK

2.1 NEUTRON MULTIPLICATION FACTOR

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95 as recommended in ANSI 57.2-1983 and in "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

The following are the conditions that are assumed in meeting this design basis for the Zion Nuclear Units 1 and 2 spent fuel storage racks.

2.2 NORMAL STORAGE

a. The fuel assembly contains the highest enrichment authorized without any control rods or any burnable poisons and is at its most reactive point in life. Even though both 15x15 Westinghouse standard and optimized (OFA) fuel may be stored, the OFA fuel is modeled since it is the most reactive of the 15x15 designs. The 15x15 OFA fuel parameters are given in Table 2-1 and a diagram of the assembly is shown in Figure 2-1. The enrichment of the 15 x 15 Westinghouse OFA fuel assembly is 3.7 w/o U-235 with no depletion or fission product buildup. The assembly is conservatively modeled with water replacing the assembly grid volume and no U-234 and U-236 in the fuel pellet.

b. The moderator is pure water at the temperature within the design limits of the pool which yields the largest reactivity. A conservative value of 1.0 gm/cm^3 is used for the density of water. No dissolved boron is included in the water in this analysis for conservatism.

c. The nominal case calculation is infinite in lateral and axial extent. Calculations show that the finite rack surrounded by a water reflector is less reactive than the nominal case infinite rack. Therefore, the nominal case of an infinite array of cells is a conservative assumption.

d. Mechanical uncertainties and biases due to mechanical tolerances during construction are treated by using "worst case" conditions. The items included in the analysis are:

-- stainless steel thickness
-- cell ID
-- center-to-center spacing
-- poison thickness
-- asymmetric assembly position

The calculation method uncertainty and bias is discussed in Section 2.4.

e. Credit is taken for the neutron absorption in full length structural materials and in solid materials added specifically for neutron absorption. The minimum poison loading (0.02 gm - B10/cm^2) is used in the poisoned cell walls.

f. A bias is included in the reactivity calculation to account for the B_4C particle self shielding.

These assumptions are summarized in Table 2-3.

2.3 POSTULATED ACCIDENTS

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more than eight inches of water separating it from the active fuel in the rest of the rack which precludes interaction).

However, accidents can be postulated which would increase reactivity such as inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. Therefore, for accident conditions, the double contingency principle of ANS N16.1-1975 is applied. This states that it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron (2000 ppm) in the storage pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

The presence of the approximately 2000 ppm boron in the pool water will decrease reactivity by more than 30% Δk. Thus $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

For normal fuel storage applications in the spent fuel pool, water is present. However, accidental criticality when fuel assemblies are stored in the dry condition is also accounted for. For this case, possible sources of moderation, such as those that could arise during fire fighting operations, are included in the analysis.

This "optimum moderation" accident is not a problem in poisoned fuel storage racks. The presence of poison plates removes the conditions necessary for "optimum moderation" so that K_{eff} continually decreases as moderator density decreases from 1.0 gm/cm^3 to 0.0 gm/cm^3 in poison rack designs. Figure 2-2 shows the behavior of K_{eff} as a function of moderator density for a typical PWR poisoned spent fuel storage rack.

2.4 METHOD FOR CRITICALITY ANALYSIS

The calculation method and cross-section values are verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. This benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

The design method which ensures the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system of codes[1,2] for cross-section generation and KENO IV[3] for reactivity determination.

The 218 energy group cross-section library[1] that is the common starting point for all cross-sections used for the benchmarks and the storage rack is generated from ENDF/B-IV data. The NITAWL[2] program adds to this library the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. The 218 groups are reduced to 19 groups by energy and spatial weighting of cross-sections using the XSDRNPM[2] program which is a one-dimensional S_n transport theory code. These multi-group cross-section sets are then used as input to KENO IV[3] which is a three-dimensional Monte Carlo theory program designed for reactivity calculations.

A set of 27 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability. The experiments range from water moderated, oxide fuel arrays separated by various materials (Boral, steel and water) that simulate LWR fuel shipping and storage conditions[4,5] to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials[6] (Plexiglas, steel and air) that demonstrate the wide range of applicability of the method.

The results and some descriptive facts about each of the 27 benchmark critical experiments are given in Table 2-2. The average K_{eff} of the benchmarks is 0.9998 which demonstrates that there is no significant bias associated with the method. The standard deviation of the bias value is 0.0014 Δk. The 95/95 one sided tolerance limit factor for 27 values is 2.26. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.0032 Δk.

The most important effect on reactivity of the mechanical tolerances is the possible reduction in the center-to-center spacing between adjacent assemblies. By stacking material and mechanical tolerances, the gap between

adjacent cells is reduced by 0.3 inches such that a minimum gap of 0.618 inches between adjacent cells is achieved. The center-to-center spacing between the fuel assemblies inside the cells can be reduced further by shifting the fuel assemblies to neighboring corners of the cells. Thus the center-to-center spacing between adjacent fuel assemblies can be reduced by a total of 0.867 inches. The KENO model of this configuration consists of an infinite array of clusters of 4 cells with a 0.618 inch gap between adjacent cells in each cluster and the fuel assemblies loaded into adjacent corners of the cells. The model guarantees that the average center-to-center cell spacing for a module of cells will be 10.35 inches. This is accomplished by increasing the gap on the opposite side of the cell by the amount it is decreased on the other side. An analysis shows that the cell center-to-center spacing reduction in combination with asymmetric fuel assembly positioning yields the highest K_{eff}.

The final result of the uncertainty analysis is that the criticality design criteria are met when the calculated effective multiplication factor, plus the total uncertainty and any biases, is less than 0.95.

These methods conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", Section 5.7, Fuel Handling System; ANSI 57.2-1983, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations", Section 6.4.2; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety"; NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage"; and the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

2.5 CRITICALITY ANALYSIS FOR RACK DESIGN

The spent fuel storage cell is shown in Figure 2-3. The sensitivities of storage lattice K_{eff} to U-235 enrichment of the fuel assembly, the storage lattice pitch, and B-10 loading in the poison plates as requested by the NRC are given in Figures 2-4 to 2-6, respectively. The error bars shown on

Figures 2-4 to 2-6 represent a 95 percent probability, 95 percent confidence level interval for the individual KENO calculations.

For normal conditions and using the method in the above sections the nominal K_{eff} is 0.9371 $\pm$ 0.0033(1σ). The maximum K_{eff} for the rack, for normal conditions, is determined in the following manner.

$$K_{eff} = K_{worst} + B_{method} + B_{part} + [(ks)^2_{worst} + (ks)^2_{method}]^{1/2}$$

Where:

K_{worst} = worst case KENO K_{eff} that includes asymmetric fuel assembly position, material tolerances, and mechanical tolerances which can result in spacings between assemblies less than nominal

B_{method} = method bias determined from benchmark critical comparisons

B_{part} = bias to account for poison particle self-shielding

ks_{worst} = 95/95 uncertainty in the worst case KENO K_{eff}

ks_{method} = 95/95 uncertainty in the method bias

Substituting calculated values in the order listed above, the result is:

$$K_{eff} = 0.9387 + 0.0 + 0.0025 + [(0.0061)^2 + (0.0032)^2]^{1/2} = 0.9481$$

Since K_{eff} is less than 0.95 including uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality is met.

2.6 ACCEPTANCE CRITERIA FOR CRITICALITY

The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, including all uncertainties, under all conditions.

3.0 CRITICALITY ANALYSIS FOR ZION NEW FUEL RACK

3.1 NEUTRON MULTIPLICATION FACTOR

Criticality of fuel assemblies in the new fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies to take advantage of neutron absorption in water and stainless steel.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95 as recommended in ANSI 57.3-1983.

The following are the conditions that are assumed in meeting this design basis for the Zion Nuclear Units 1 and 2 new fuel storage racks.

3.2 NORMAL STORAGE

a. The fuel assembly contains the highest enrichment authorized without any control rods or any burnable poisons and is at its most reactive point in life. Even though both 15x15 Westinghouse standard and optimized (OFA) fuel may be stored, the OFA fuel is modeled since it is the most reactive of the 15x15 designs. The enrichment of the 15 x 15 Westinghouse OFA fuel assembly is 4.0 w/o U-235 with no depletion or fission product buildup. The assembly is conservatively modeled with the assembly grid volume removed and no U-234 and U-236 in the fuel pellet.

b. The nominal case calculation is infinite in lateral and axial extent. Calculations show that the finite rack is less reactive than the nominal case infinite rack. Therefore, the nominal case of an infinite array of cells is a conservative assumption.

c. Even though the fresh fuel is stored dry, the nominal calculation conservatively assumes the storage array is flooded with water at a density of 1 gm/cm^3. No dissolved boron is included in the water.

d. Mechanical uncertainties and biases due to mechanical tolerances during construction are treated by using "worst case" conditions. The items included in the analysis are:

- angle iron thickness
- cell ID
- center-to-center spacing
- asymmetric assembly position

The calculation method uncertainty and bias is discussed in Section 3.4.

e. Credit is taken for the neutron absorption in full length stainless steel structural material.

These assumptions are summarized in Table 3-1.

3.3 POSTULATED ACCIDENTS

Most accident conditions will not result in an increase in K_{eff} of the rack. An example is the dropping of a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more than eight inches separating it from the active fuel in the rest of the rack which precludes interaction).

However, accidents can be postulated (under flooded conditions) which would increase reactivity such as inadvertent drop of an assembly between the outside periphery of the rack and vault wall. Therefore, for accident conditions, the double contingency principle of ANS N16.1-1975 is applied. This states that it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the absence of water in the storage vault can be assumed as a realistic initial condition since assuming its presence would be a second unlikely event.

Because the most limiting accident is the introduction of moderation into the storage vault, this accident is considered in determining the maximum K_{eff} for the fresh fuel storage vault. It is known that an "optimum moderation" phenomenon occurs at low water densities where K_{eff} may be comparable to or higher than K_{eff} at a water density of 1 gm per cm^3. This phenomenon was investigated for the Zion fresh fuel storage vault by first computing the maximum K_{eff} based on an infinite storage array. The maximum K_{eff}, which exceeded the K_{eff} at 1.0 gm per cm^3, was found to exist at a water density in the range of .04 to 0.06 gm per cm^3.

A finite model of the actual Zion fresh fuel storage vault was then developed. A KENO calculation was performed with a water density of 0.06 gm per cm^3 and K_{eff} was found to be less tha 0.9. The angle iron surrounding each fuel assembly was not modeled thus yielding a conservative result. Use of the finite model accounts for the large amount of neutron leakage from the storage array at low water densities. This approach to analyze the "optimum moderation" phenomenon is much more realistic than the infinite array approach.

3.4 METHOD FOR CRITICALITY ANALYSIS

The calculation method, cross section library and computer codes, and experimental data base used in the spent fuel rack analysis is used for the fresh fuel rack analysis.

The most important effect on reactivity of the mechanical tolerances is the possible reduction in the center-to-center spacing between adjacent assemblies. By stacking material and mechanical tolerances, the gap between adjacent cells is reduced by 0.25 inches such that a minimum gap of 11.375 inches between adjacent cells (angle iron) is achieved. The center-to-center spacing between the fuel assemblies inside the cells can be reduced further by shifting the fuel assemblies to neighboring corners of the cells. Thus the center-to-center spacing between adjacent fuel assemblies can be reduced by a total of 0.93 inches. A decrease in the center-to-center spacing between fuel assemblies of 0.93 inches, however, is not sufficient to cause an increase in the K_{eff} from the nominal case. As mentioned in Section 3.3, a gap of eight

inches or more between fuel assemblies will preclude interaction. Thus, the K_{eff} from the nominal case will be used to determine the maximum K_{eff} for normal operation.

The final result of the uncertainty analysis is that the criticality design criteria are met when the calculated effective multiplication factor, plus the total uncertainty and any biases, is less than 0.95.

These methods conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", Section 5.7, Fuel Handling System; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety", ANSI 57.3-1983, "Design Requirements for New Fuel Storage Facilities at LWR Plants", Section 6.2.4.

3.5 CRITICALITY ANALYSIS FOR RACK DESIGN

The fresh fuel storage cell is shown in Figure 3-1. The sensitivity of storage lattice K_{eff} to U-235 enrichment of the fuel assembly and the storage lattice pitch is given in Figures 3-2 and 3-3, respectively. The error bars shown in Figures 3-2 and 3-3 represent a 95 percent probability, 95 percent confidence level interval for the individual KENO calculations.

For normal conditions and using the method in the above section, the nominal K_{eff} is 0.9010 $\pm$ .0041(1σ). The maximum K_{eff} for the rack, for normal conditions, is determined in the following manner.

$$K_{eff} = K_{nominal} + B_{method} + [(ks)^2_{nominal} + (ks)^2_{method}]^{1/2}$$

where:

$K_{nominal}$ = nominal case KENO K_{eff}

B_{method} = method bias determined from benchmark critical comparisons

$ks_{nominal}$ = 95/95 uncertainty in the nominal case KENO K_{eff}

ks_{method} = 95/95 uncertainty in the method bias

Substituting calculated values in the order listed above, the result is:

$$K_{eff} = 0.9010 + 0.0 + [(.0069)^2 + (.0032)^2]^{1/2} = .9086$$

Since K_{eff} is less than 0.95 including uncertainties at a 95/95 probability/ confidence level, the acceptance criteria for criticality is met.

REFERENCES

1. W.E. Ford III, et al, "A 218-Group Neutron Cross-Section Library in the AMPX Master Interface Format for Criticality Safety Studies," ORNL/CSD/TM-4 (July 1976).

2. N.M. Greene, et al, "AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL/TM-3706 (March 1976).

3. L.M. Petrie and N.F. Cross, "KENO IV-An Improved Monte Carlo Criticality Program," ORNL-4938 (November 1978).

4. S.R. Bierman, et al, "Critical Separation Between Subcritical Clusters of 2.35 wt % $^{235}UO_2$ Enriches UO_2 Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories PNL-2438 (October 1977).

5. S.R. Bierman, et al, "Critical Separation Between Subcritical Clusters of 4.29 wt % $^{235}UO_2$ Enriched UO_2 Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories PNL-2614 (March 1978).

6. J. T. Thomas, "Critical Three-Dimensional Arrays of U (93.2) - Metal Cylinders," Nuclear Science and Engineering, Volume 52, pages 350-359 (1973).

Table 2-1

Westinghouse 15x15 OFA Fuel Specifications

Fuel Material	UO_2
U^{235} Enrichment	Variable
UO_2 Density	95% theoretical
Rod Pitch	0.5630"
Pellet O.D.	0.3659"
Pellet Dishing Fraction	1.1778%
Clad Material	Zircalloy-4
Clad I.D.	0.3734"
Clad O.D.	0.4220"
Clad Thickness	0.0243"
Guide Tube Material	Zircalloy-4
Guide Tube I.D.	0.4980"
Guide Tube O.D.	0.5320"
Guide Tube Thickness	0.0170"
Instrument Tube Material	Zircalloy-4
Instrument Tube I.D.	0.4980"
Instrument Tube O.D.	0.5320"
Instrument Tube Thickness	0.0170"
Number Fuel Rods per Assembly	204
Number Guide Tube per Assembly	20
Number Instrument Tube per Assembly	1

Table 2-2

BENCHMARK CRITICAL EXPERIMENTS[4,5,6]

	General Description	Enrichment w/o U235	Reflector	Separating Material	Characterizing Separation (cm)	K_{eff}*
1.	UO_2 rod lattice	2.35	water	water	11.92	1.004 ± .004
2.	UO_2 rod lattice	2.35	water	water	8.39	0.993 ± .004
3.	UO_2 rod lattice	2.35	water	water	6.39	1.005 ± .004
4.	UO_2 rod lattice	2.35	water	water	4.46	0.994 ± .004
5.	UO_2 rod lattice	2.35	water	stainless steel	10.44	1.005 ± .004
6.	UO_2 rod lattice	2.35	water	stainless steel	11.47	0.992 ± .004
7.	UO_2 rod lattice	2.35	water	stainless steel	7.76	0.992 ± .004
8.	UO_2 rod lattice	2.35	water	stainless steel	7.42	1.004 ± .004
9.	UO_2 rod lattice	2.35	water	boral	6.34	1.005 ± .004
10.	UO_2 rod lattice	2.35	water	boral	9.03	0.992 ± .004
11.	UO_2 rod lattice	2.35	water	boral	5.05	1.001 ± .004
12.	UO_2 rod lattice	4.29	water	water	10.64	0.999 ± .005
13.	UO_2 rod lattice	4.29	water	stainless steel	9.76	0.999 ± .005
14.	UO_2 rod lattice	4.29	water	stainless steel	8.08	0.998 ± .006
15.	UO_2 rod lattice	4.29	water	boral	6.72	0.998 ± .005
16.	U metal cylinders	93.2	bare	air	15.43	0.998 ± .003
17.	U metal cylinders	93.2	paraffin	air	23.84	1.006 ± .005
18.	U metal cylinders	93.2	bare	air	19.97	1.005 ± .003
19.	U metal cylinders	93.2	paraffin	air	36.47	1.001 ± .004
20.	U metal cylinders	93.2	bare	air	13.74	1.005 ± .003
21.	U metal cylinders	93.2	paraffin	air	23.48	1.005 ± .004
22.	U metal cylinders	93.2	bare	plexiglass	15.74	1.010 ± .003
23.	U metal cylinders	93.2	paraffin	plexiglass	24.43	1.006 ± .004
24.	U metal cylinders	93.2	bare	plexiglass	21.74	0.999 ± .003
25.	U metal cylinders	93.2	paraffin	plexiglass	27.94	0.994 ± .005
26.	U metal cylinders	93.2	bare	steel	14.74	1.000 ± .003
27.	U metal cylinders	93.2	bare	plexiglass, steel	16.67	0.996 ± .003

*Rounded to three places.

~~WESTINGHOUSE PROPRIETARY CLASS 2~~

Table 2-3

Assumptions Made in Criticality Analysis of Zion Spent Fuel Storage Racks

a) 15x15 Westinghouse OFA fuel enriched to 3.7 w/o U-235 stored at its most reactive point in life. Credit for fuel depletion, fission product buildup or presence of U-234, U-236 or assembly grids is not taken.

b) Pure water at a density of 1 gm/cm^3 used.

c) Infinite array in lateral and axial extent modeled.

d) Mechanical uncertainties/tolerances arising from construction treated by using "worst case" conditions.

e) The minimum poison loading (0.02 gm - $B10/cm^2$) used.

f) A bias accounting for the B_4C particle self-shielding used.

Table 3-1

Assumptions Made in Criticality Analysis of Zion Fresh Fuel Storage Racks

a) 15x15 Westinghouse OFA fuel enriched to 4.0 w/o U-235 stored at its most reactive point in life. Credit for fuel depletion, fission product buildup or presence of U-234, U-236 or assembly grids is not taken.

b) Pure water at a density of 1 gm/cm^3 used for normal conditions and 0.06 g/cm^3 used for "optimum moderation".

c) Infinite array in lateral and axial extent modeled for normal conditions and finite array in all directions modeled for "optimum moderation".

d) Mechanical uncertainties/tolerances arising from construction treated by using "worst case" conditions.

e) Credit taken for presence of stainless steel structural material for normal conditions but not for "optimum moderation".

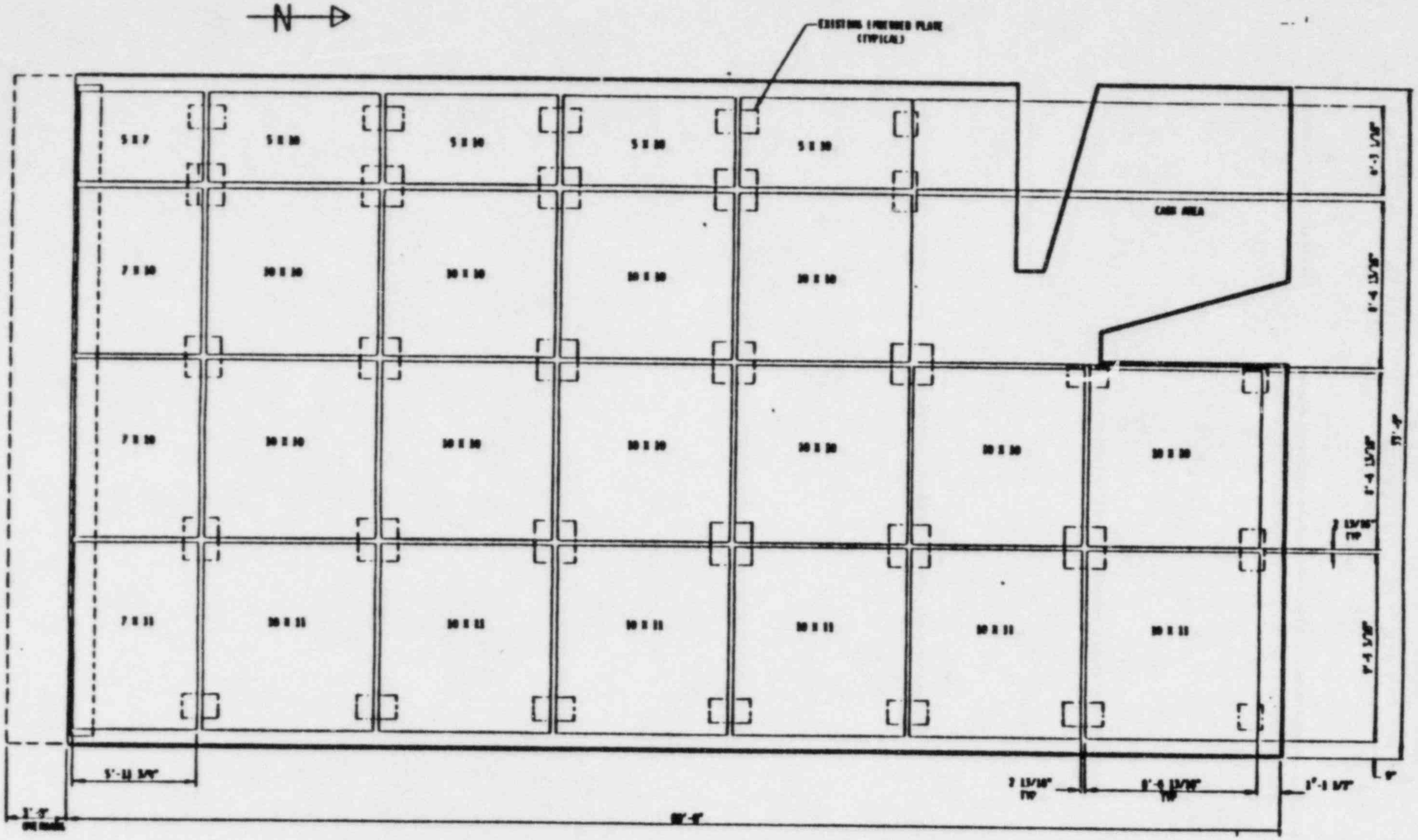


FIGURE 1-1 ZION SPENT FUEL POOL ARRANGEMENT (MODIFIED)

6 - 10 X 11 RACKS = 660 SPACES
10 - 10 X 10 RACKS = 1000 SPACES
1 - 7 X 11 RACKS = 77 SPACES
2 - 7 X 10 RACKS = 140 SPACES
1 - 5 X 7 RACKS = 35 SPACES
4 - 5 X 10 RACKS = 200 SPACES

TOTAL = 2112 SPACES

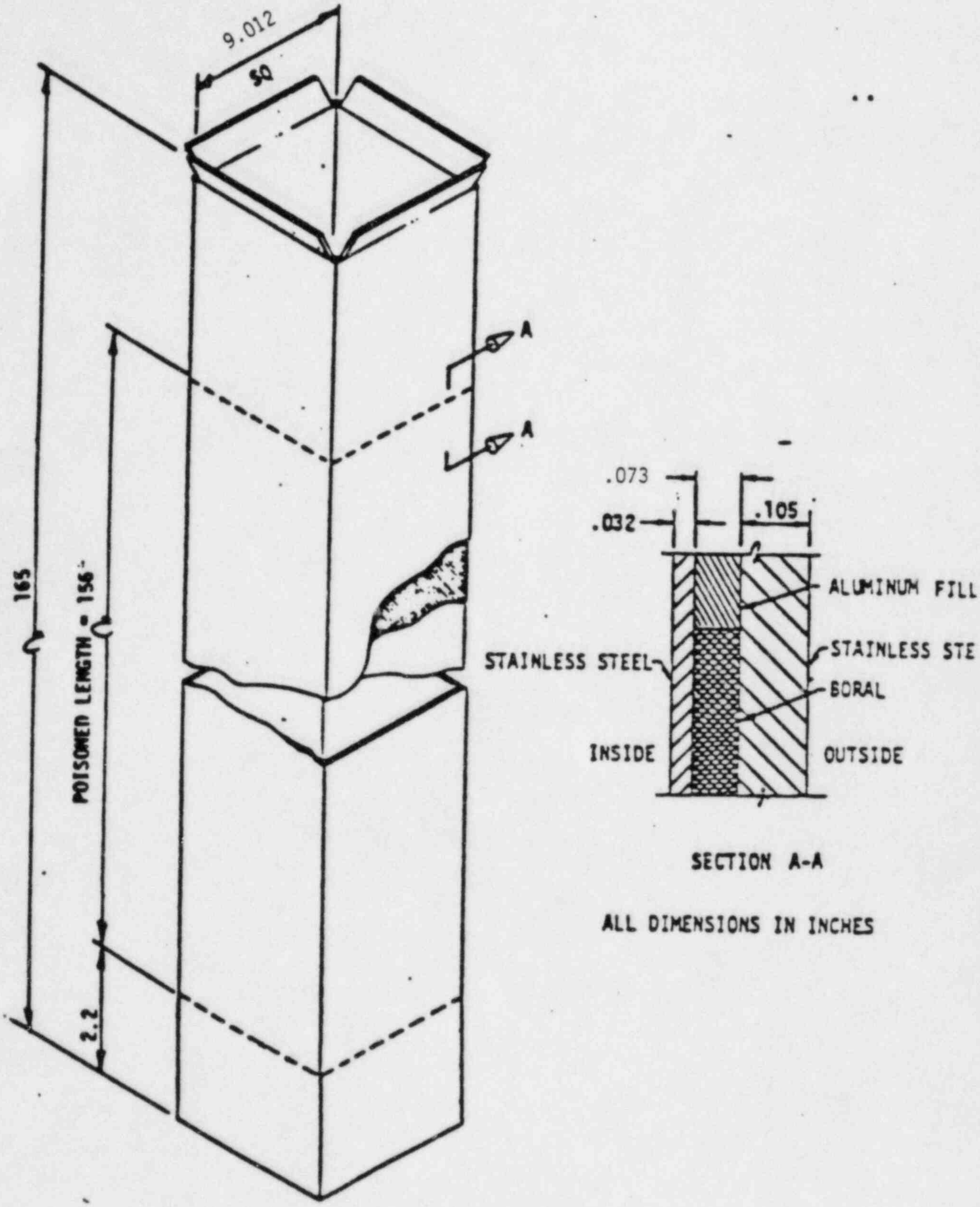


FIGURE 1-2 STAINLESS STEEL TUBE WITH BORAL CORE FOR ZION FUEL STORAGE

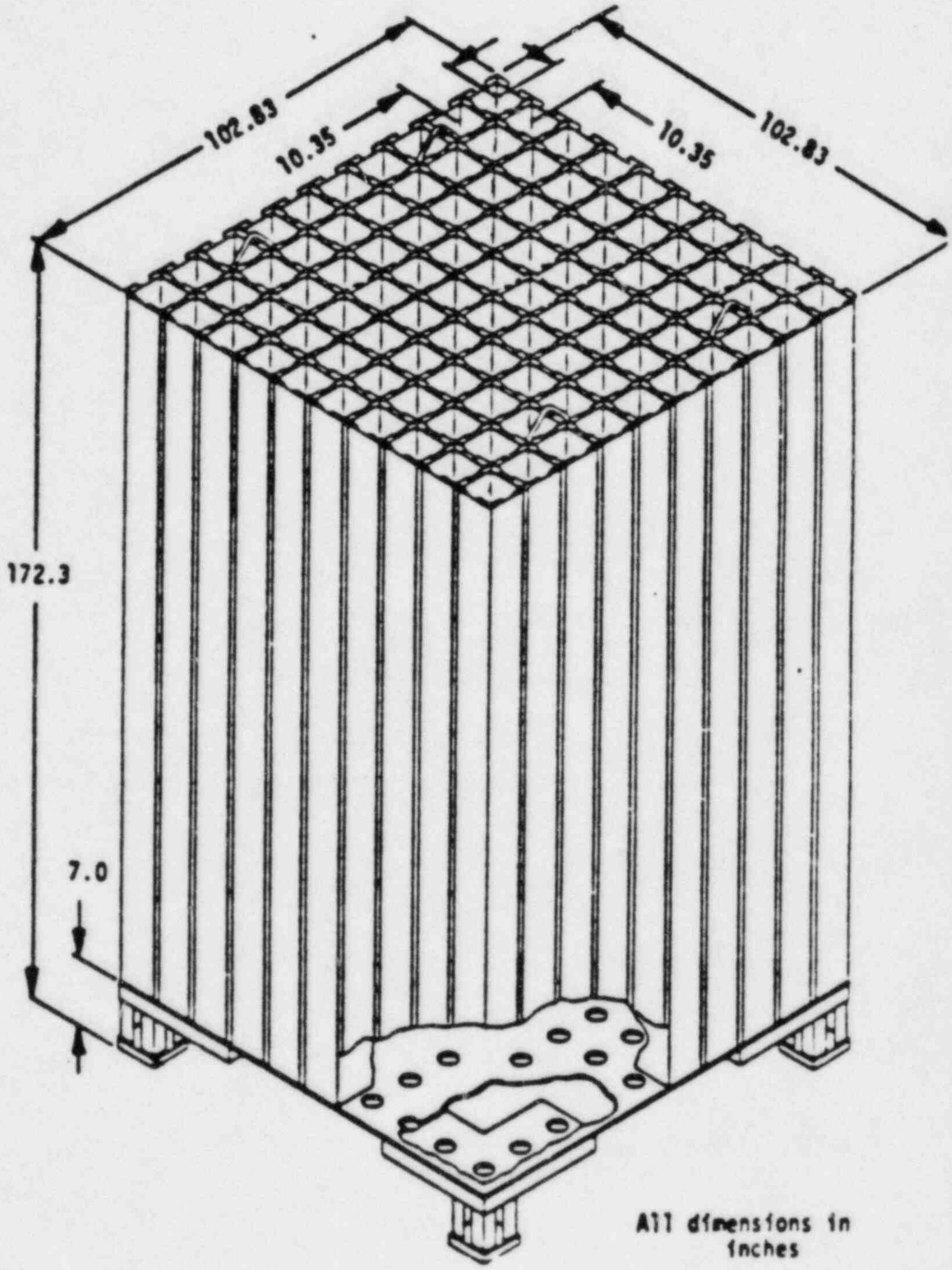
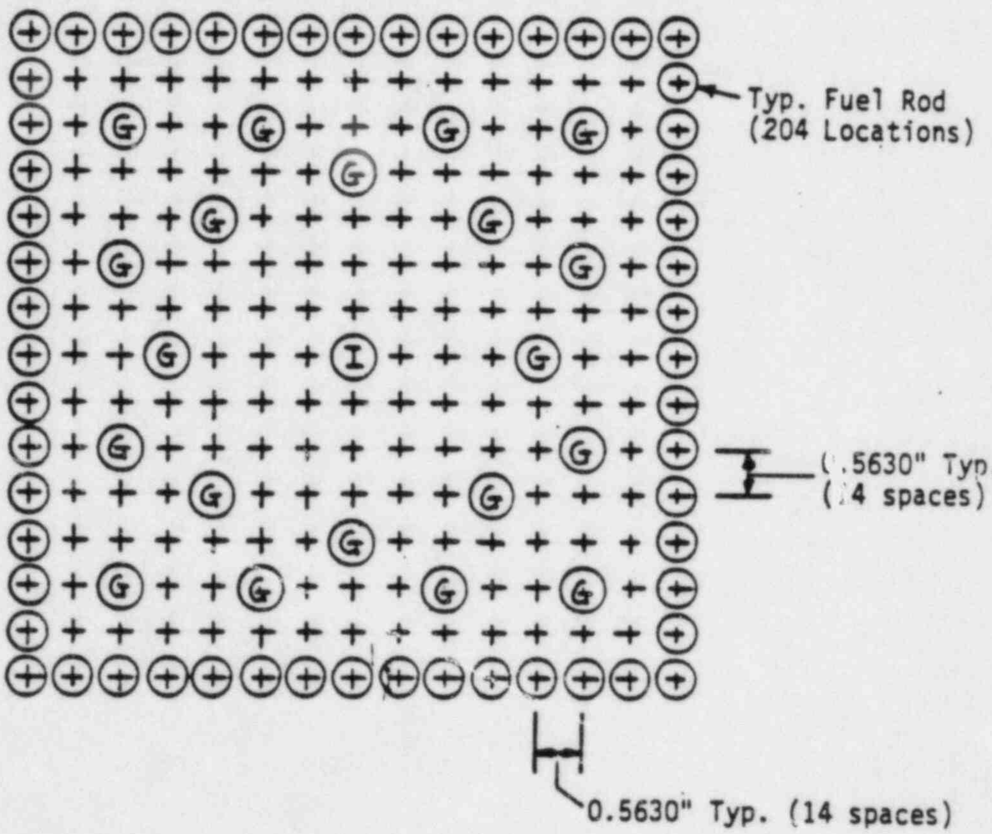


FIGURE 1-3 10 X 10 SPENT FUEL RACK
ZION UNITS 1 AND 2

Figure 2-1

Westinghouse 15x15 OFA Fuel Assembly



Legend

G Control Rod Guide Tube (20 Locations)

I Instrumentation Thimble Guide Tube (1 Location)

FIGURE 2-2 K_{eff} VS. WATER MODERATOR

FOR A TYPICAL "POISONED" SPENT FUEL STORAGE RACK

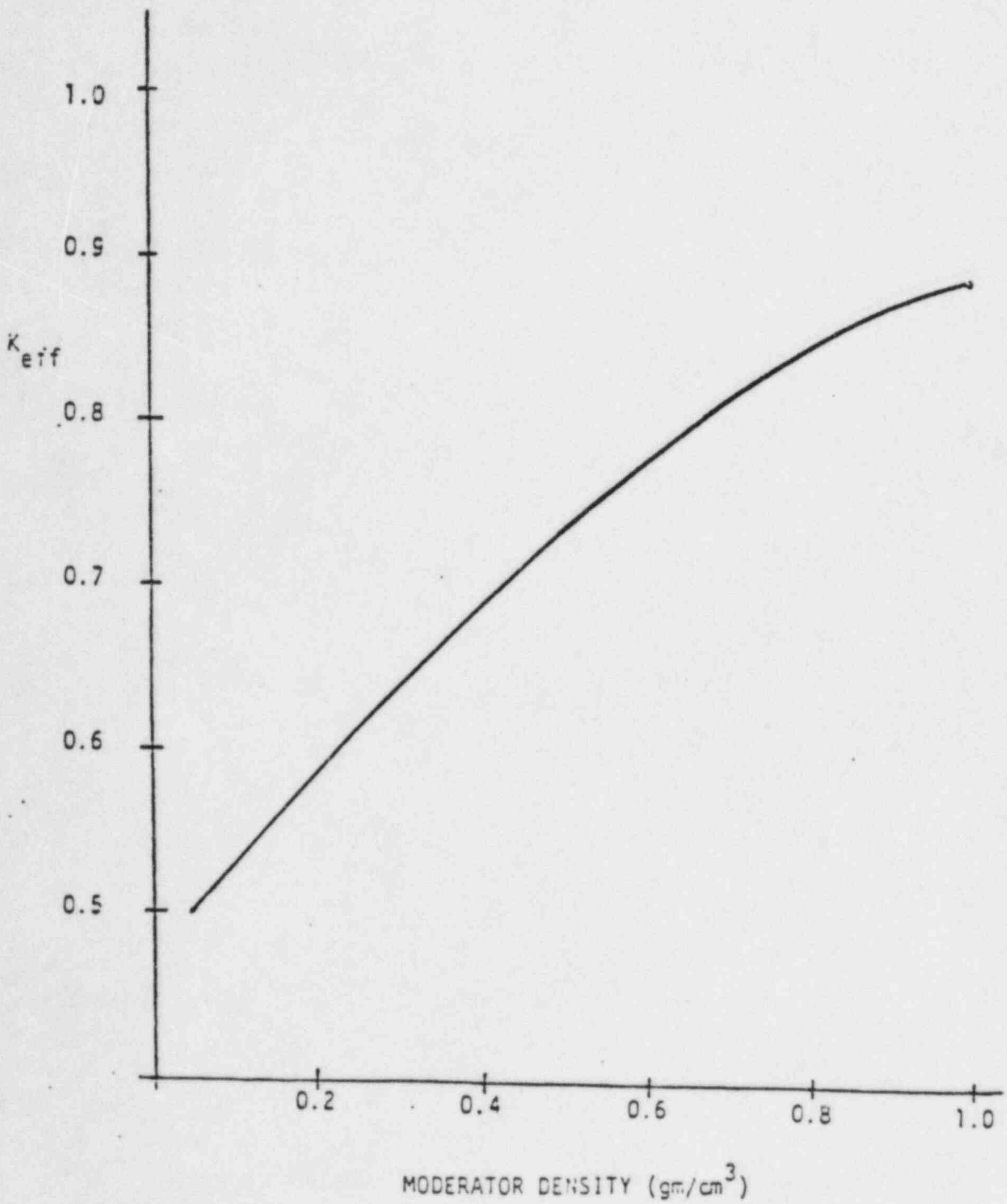


Figure 2-3

Nominal Dimensions of Zion Poisoned Spent Fuel Storage Racks

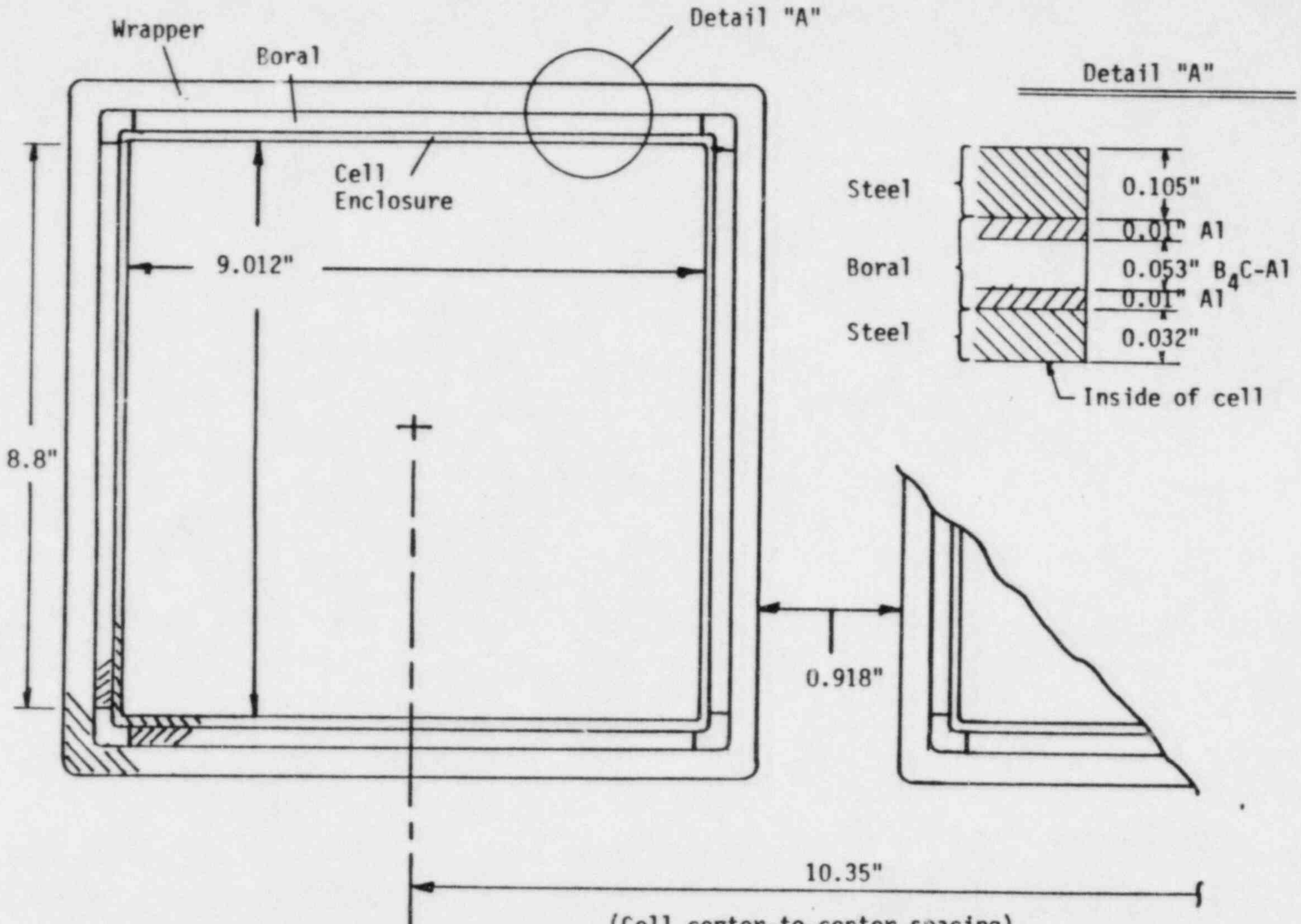
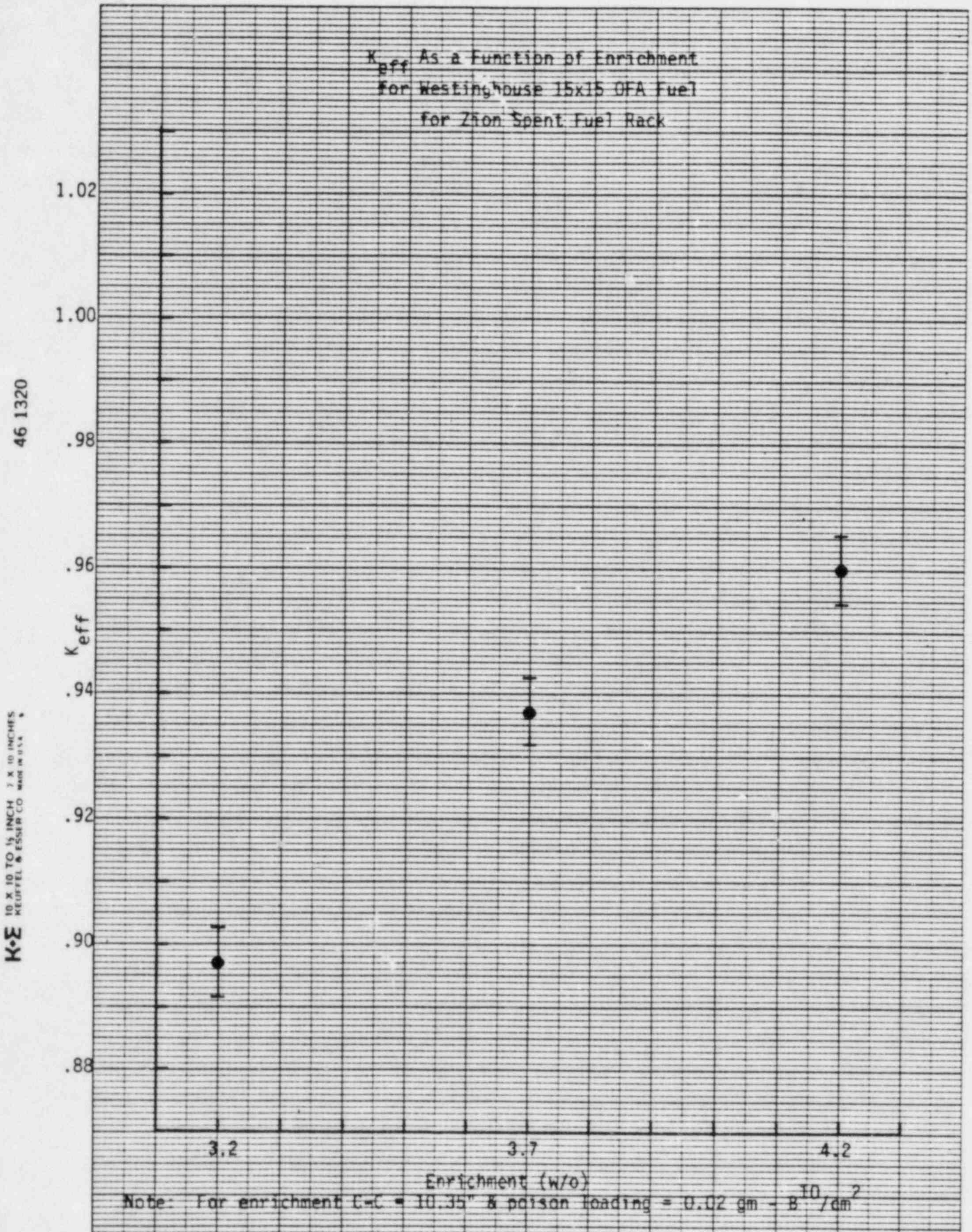
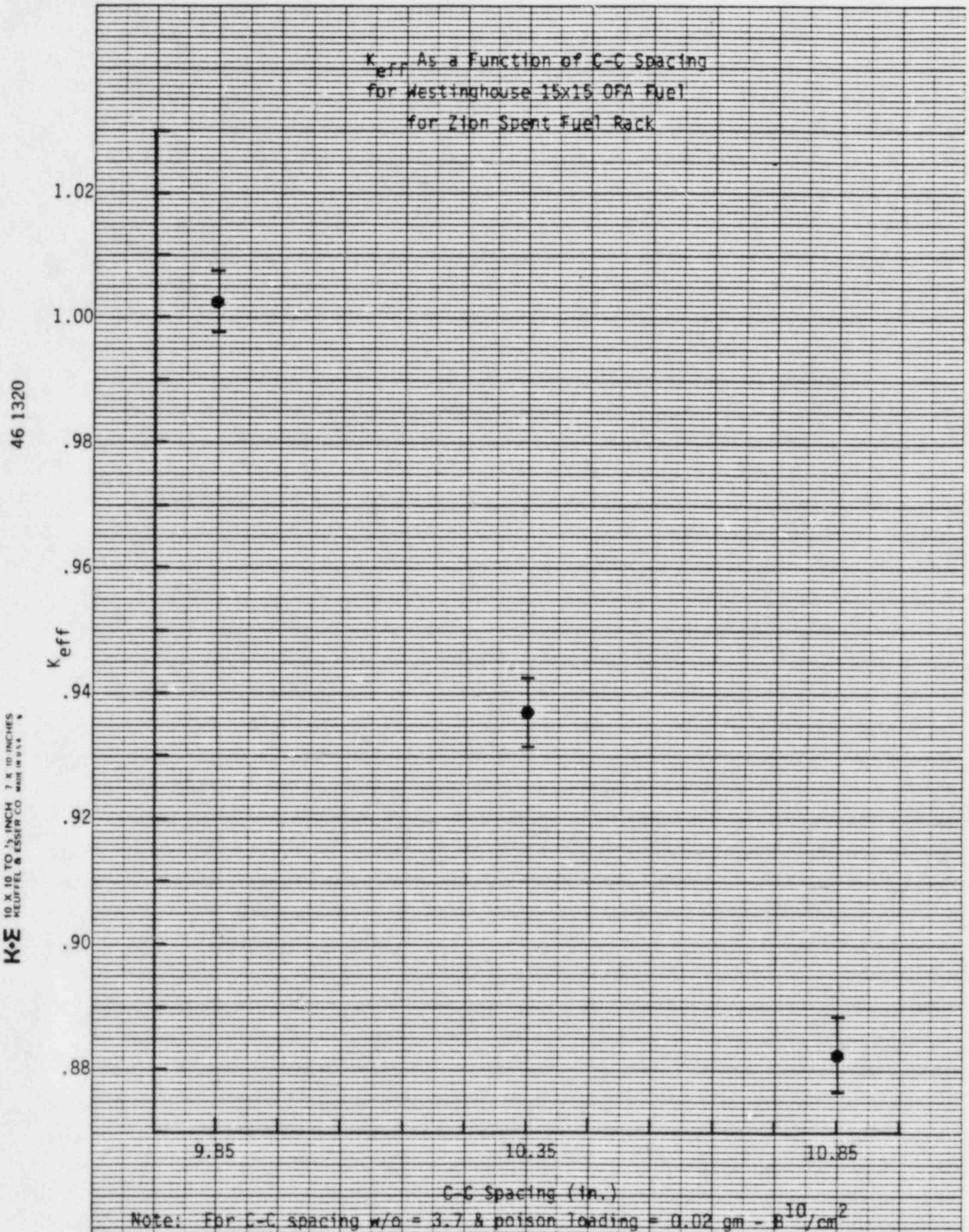


Figure 2-4



K_{eff} As a Function of Enrichment
for Westinghouse 15x15 OFA Fuel
for Zion Spent Fuel Rack
1.02
1.00
.98
.96
.94
.92
.90
.88
K_{eff}
3.2
3.7
4.2
Enrichment (W/o)
Note: For enrichment C-C = 10.35" & poison loading = 0.02 gm - B^{10}/cm^2
K•E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.
46 1320

Figure 2-5



K_{eff} As a Function of C-C Spacing
for Westinghouse 15X15 OFA Fuel
for Zion Spent Fuel Rack
1.02
1.00
.98
.96
.94
.92
.90
.88
K_{eff}
9.85
10.35
10.85
C-C Spacing (in.)
Note: For C-C spacing w/o = 3.7 & poison loading = 0.02 gm - B^{10}/cm^2
46 1320
K•E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

Figure 2-6

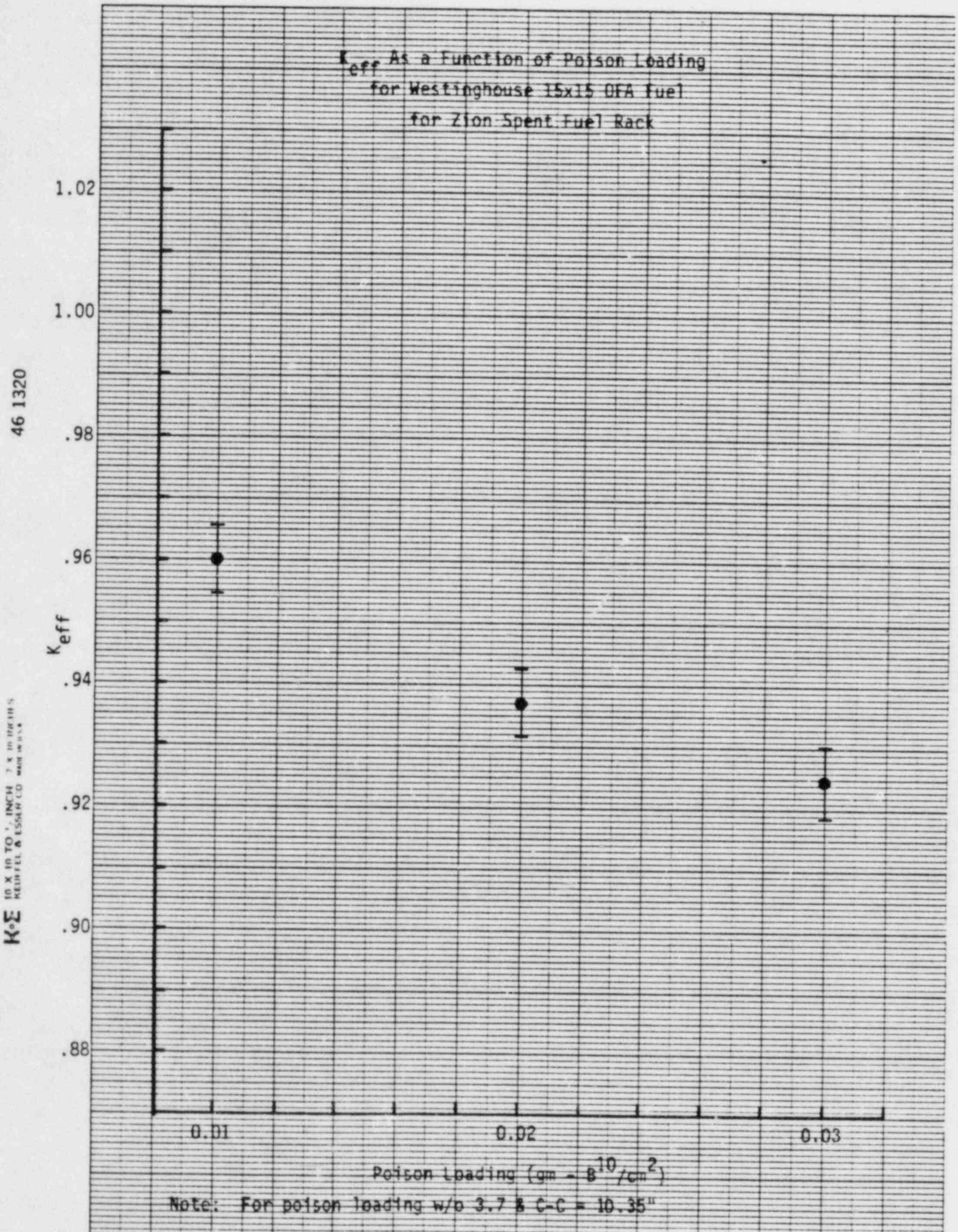

K_{eff} As a Function of Poison Loading
for Westinghouse 15x15 OFA Fuel
for Zion Spent Fuel Rack
1.02
1.00
.98
.96
.94
.92
.90
.88
K_{eff}
0.01
0.02
0.03
Poison Loading (gm - B^{10}/cm^2)
Note: For poison loading w/o 3.7 & C-C = 10.35"
46 1320
K&E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

Figure 3-1

Nominal Dimensions of Zion Fresh

Fuel Storage Racks

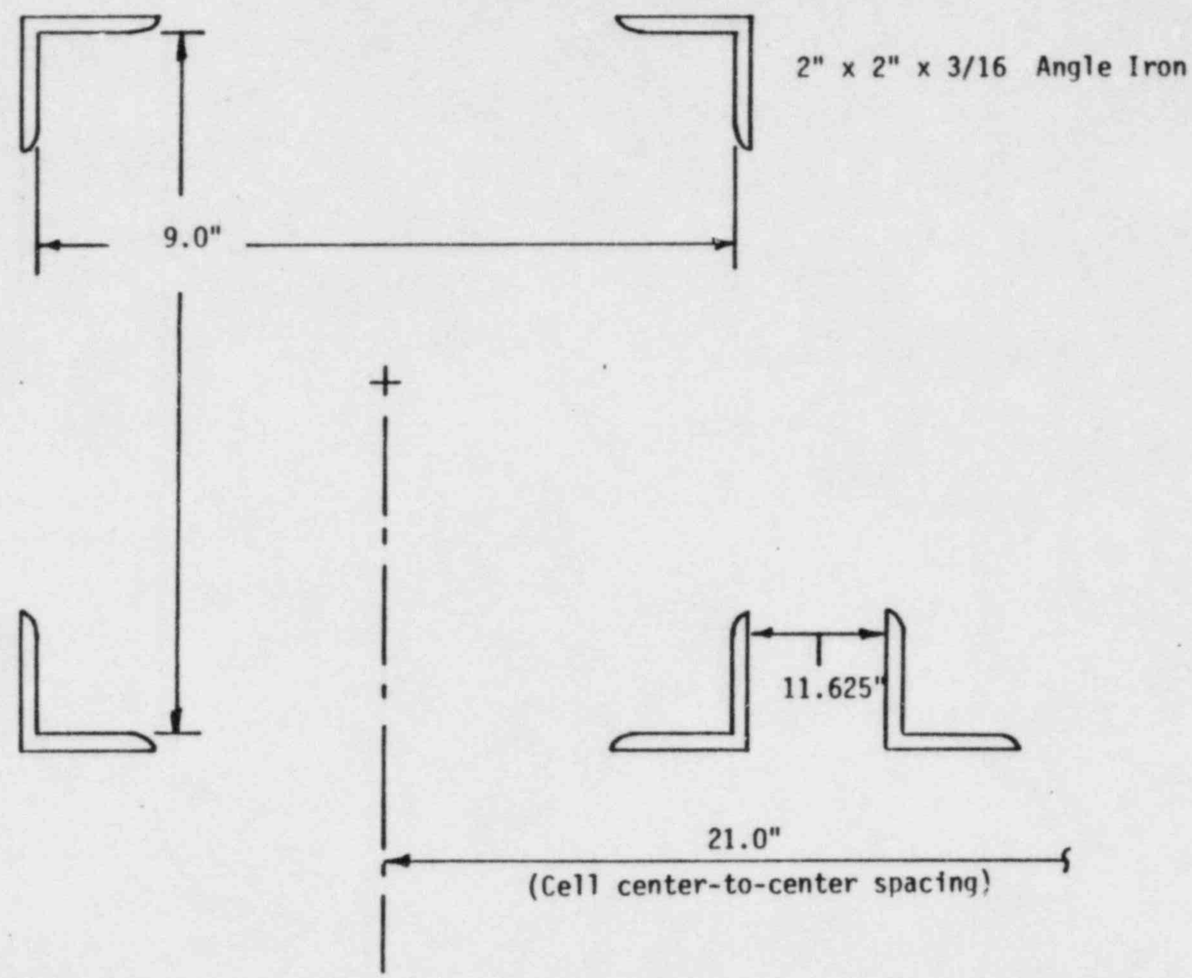


Figure 3-2

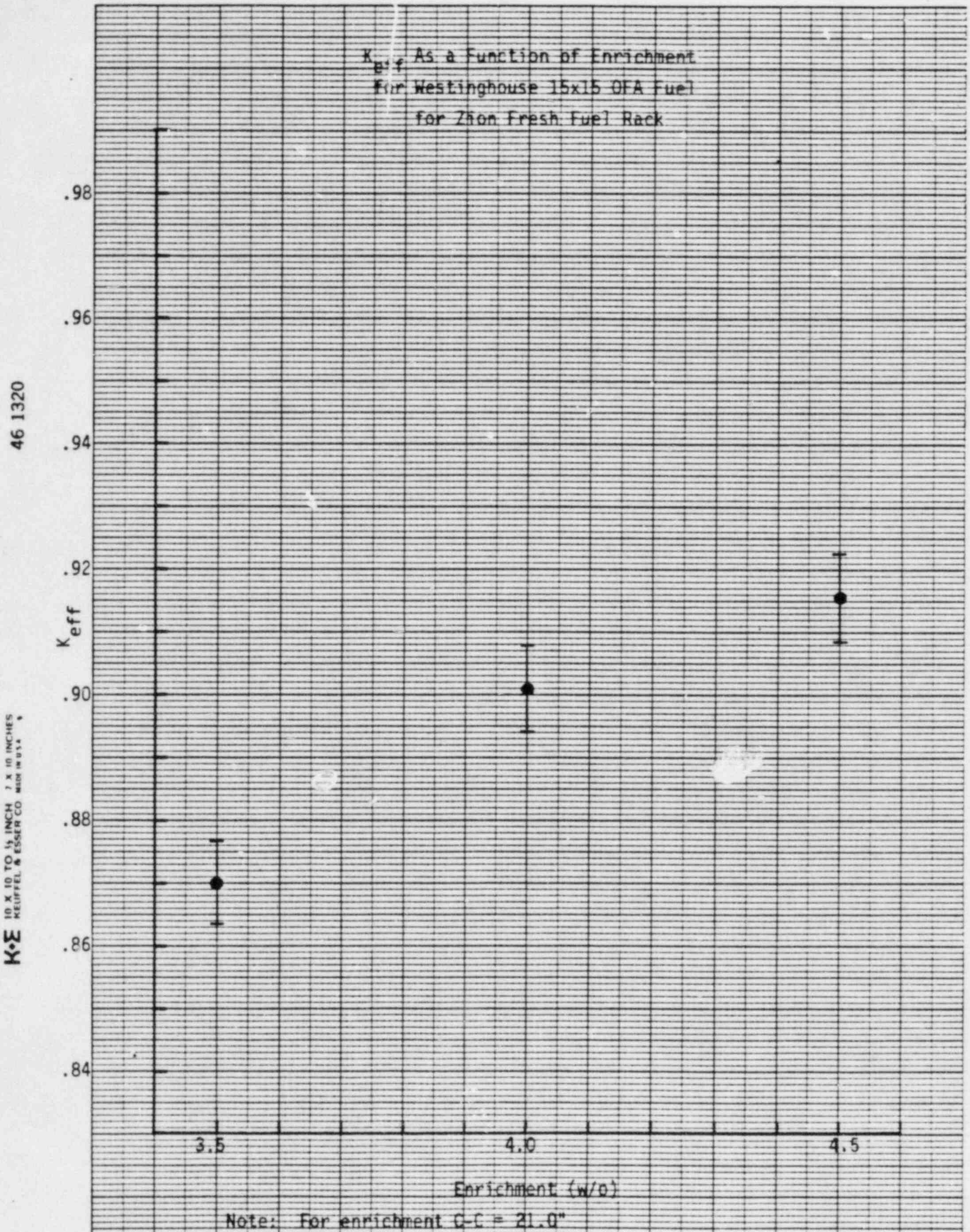
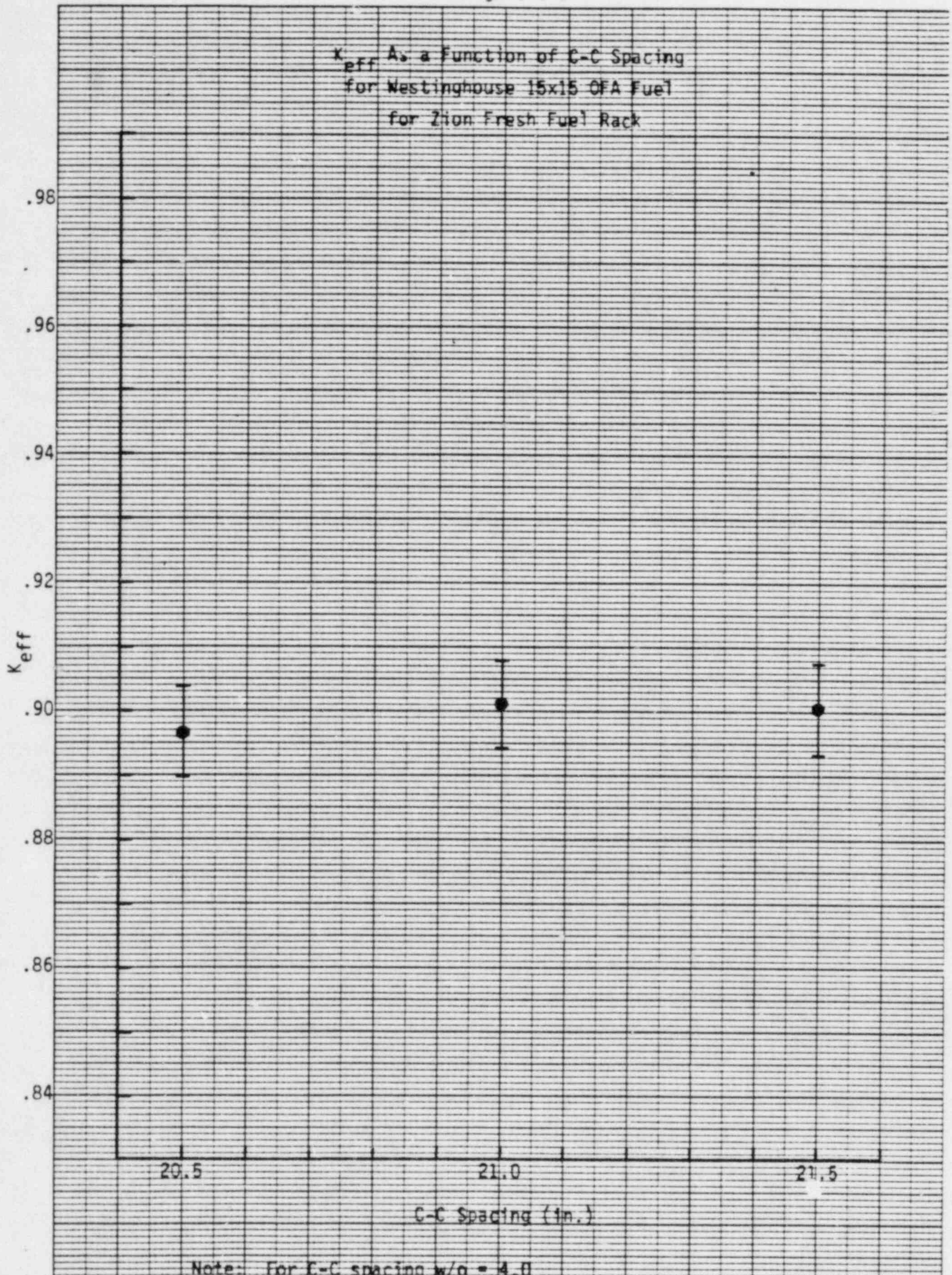

K$_{eff}$ As a Function of Enrichment
For Westinghouse 15x15 OFA Fuel
for Zion Fresh Fuel Rack
.98
.96
.94
.92
.90
.88
.86
.84
K$_{eff}$
3.5
4.0
4.5
Enrichment (w/o)
Note: For enrichment C-C = 21.0"
46 1320
K•E 10 X 10 TO ½ INCH 7 X 10 INCHES
KEUFFEL & ESSER CO.

Figure 3-3



K_{eff} As a Function of C-C Spacing
for Westinghouse 15x15 OFA Fuel
for Zion Fresh Fuel Rack
.98
.96
.94
.92
.90
.88
.86
.84
K_{eff}
20.5
21.0
21.5
C-C Spacing (in.)
Note: For C-C spacing w/o = 4.0