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U. S. Nuclear Regulatory Commission
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License SNM-1067
Docket 70-1100

October 17, 1979

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. L. C. Rouse, Chief
Fuel Processing & Fabrication Branch
Division of Fuel Cycle & Material Safety

Reference: Amendment Application dated April 23, 1979 and Supplements dated June 27, 1979 and August 7, 1979

Gentlemen:

In our original amendment application dated April 23, 1979 it was stated that an independent review of all criticality safety calculations would be performed by the Nuclear Safety Committee. That review, including independent analysis, has been completed and is hereby submitted as supplementary information to our original application.

One section of our present license allows storage of touching clad rods in horizontal storage packed in a hexagonal lattice to a maximum slab thickness of 15 inches. This section was inadvertently omitted in our amendment application and is included in this transmittal. It is requested that Page C-17 Revision 1 dated 7/16/79 be replaced by the corresponding attached page (Revision 2) dated 10/8/79.

If you have any questions regarding this application, please contact Mr. G. J. Bakevich of my staff on extension 3150.

Very truly yours,

H V Lichtenberger

H. V. Lichtenberger
Vice President-Nuclear Fuel
Nuclear Power Systems-Manufacturing

HVL/GJB/ssb
Enclosures

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diameter pellets is 6.7". Applying a safety factor of 1.2 yields a slab limit of 5.5 inches.

6.3 Close Packed Rods

Touching clad rods in horizongal storage packed in a hexagonal lattice
* have been analyzed as reported in Figure II-5 of WCAP 2999. For $k_e = 0.99$,
* the slab thickness with full water reflection is in excess of 19 inches, and an allowable slab limit of 15 inches will be applied.

6.4 Transfer of Material

Material may be transferred on carts which accommodate one mass or slab limited SIU, or may be transferred by hand, one SIU at a time. Carts used for mass limited SIU's shall provide for centering of the unit, and shall measure at least three feet on a side.

Because most spacing areas do not extend beyond the physical boundary of the equipment, spacing between transfer carts and the equipment is of no concern. In cases where the spacing area extends beyond the equipment boundaries, such as the storage facilities, the spacing boundary will be indicated with colored tape. The tape may be crossed by carts only when they contain no more than one mass or volume limited SIU, and then only to permit an operator to transfer that SIU to an available storage position.

Interoffice Correspondence

POWER SYSTEMS

H. V. Lichtenberger

J. R. Dietrich

October 15, 1979

Subject: Review of Criticality Assessment in Amendment Application for License SNM 1067

The Nuclear Safety Committee, through a subcommittee consisting of NSC members R. L. Hellens, S. Visner, and J. R. Dietrich, has directed a review of the criticality safety calculations contained in the recently proposed amendment to the Combustion Engineering Material License No. SNM-1067 Docket 70-1100 (submitted 4-23-79, with additions submitted 6-27-79 and 8-7-79). This proposed amendment raises the fuel enrichment limit to 4.1 wt % U-235. Check calculations for selected cases and other analyses were made for the subcommittee by R. S. Harding, R. J. Klc.z, and L. C. Noderer, of the Nuclear Engineering Department, and the results are contained in the attached memorandum. The calculations described in the memorandum were entirely independent of those contained in the license amendment application except for the generation of few-group cross sections, a portion of the analysis that was originally performed in the Nuclear Engineering Department for the calculations contained in the license amendment application. The end results of the calculations employing these cross sections were, however, examined critically by comparison with results of similar calculations made at other times.

The subcommittee of the Nuclear Safety Committee has gone over the attached memorandum in detail with its authors, reviewing both the methods used and the implications of the results relative to those reported in the license amendment application. The review and the associated analyses have focused on three areas encompassing the likely potential sources of error which, in turn, might affect the validity of the criticality safety calculations.

1. Independent evaluation of those situations in which Safe Individual Unit (SIU) limits are used.

2. Verification that geometry and materials are correctly represented in Monte Carlo calculations.

3. Evaluation of the error inherent in few-group calculations for moderated configurations.

The judgement of the subcommittee was that independent computer calculations of most of the specific situations cited in the amendment were not required because the codes employed had already been subjected to benchmark checks reported in Exhibit D of the license amendment. The parametric s...dy (item 3, above) did, however, require a series of new calculations performed by the Nuclear Engineering Department of the C-E Nuclear Power Systems Division using a variety of design codes.

The results of the review revealed no errors in the original calculations provided in the amendment which would significantly influence conclusions about the safety case for the facility at the new enrichment limits.

In particular, no disagreements were found with the proposed posting of work stations based on SIU limits. This is not surprising since the eight cases treated in this way are relatively simple configurations and there is not much room for disagreement.

Differences did appear between the original and the reviewers' construction of some of the input to the twelve Monte Carlo calculations, and in those cases in which the difference appeared to be potentially important, the computer input was changed and the cases rerun. The changes are cited in the attached memorandum. In short, it appears that uncertainties due to interpretation of geometry and material content were found to be less than 1.5% in reactivity, an uncertainty sufficiently smaller than the level of subcriticality that it does not reduce the margin required for safe operation to a significant extent.

The calculations for most of the fuel rod configurations dealt with in the amendment used the "few-group" methods normally employed in core design work to represent the effects of moderation by water mist and flooding conditions. These methods were developed to represent large regions of relatively uniform neutron spectrum typical of power reactor cores, and they are known to lose accuracy when applied to situations in which the neutron spectrum either varies rapidly in space or is determined by strong interaction between two or more dissimilar regions. Two examples of this sort are discussed in the attached memorandum to show the systematic trend of the multiplication constant with numbers of groups used in the spatial calculations and the dependence of the magnitude of the error on the physical configuration of the case. The cases represent infinite square arrays of fuel assemblies, moderated by mist, but arranged with differing interassembly gaps. The example representative of the facility arrangement shows a non-conservative error in the four-group calculation of about 3% Δ k/k for a situation which is roughly 20% subcritical. This error is judged by the subcommittee to be well within the expected and tolerable range for analyses of this sort.

The second example provides a more severe test of the four-group analysis; it exhibits an error in the range of 7-8% Δ k/k but it is not representative of fuel assembly arrays permitted in the manufacturing facility by the license. It does serve to indicate, however, that the errors associated with few-group methods can be markedly reduced by using more advanced cross-section generation methods for few-group constants which take better account of spectrum interaction effects. These advanced methods are just coming into use at C-E and will be applied to the NSC review function as occasions arise. Although they help to understand error trends, it is not felt that extensive analyses of this sort are needed to support the calculations presented in the current license amendment.

The subcommittee concludes from this review that: (1) the calculations in the license amendment application have been carried out with adequate understanding of the methods involved; and (2) although the calculational results are subject to some uncertainty, the calculated margins to criticality are adequate to cover the uncertainties. The review has developed information useful in the assessment of uncertainties due to limitations of computer codes. Beyond that it has found only minor discrepancies in the original calculations: a drawing error and an error in taking information from a drawing. These have proved to be inconsequential and the subcommittee finds no fault in the criticality safety argument presented in the amendment application.

J. R. Dietrich

J. R. Dietrich
Chairman, Nuclear
Safety Committee

JRD:jd
Att.

cc w/att. - B

F. J. Pianki
G. J. Bakevich

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Report to Nuclear Safety Committee on
Review of Nuclear Fuel Manufacturing
Application For An Increase In Limiting
UO_2 Enrichment From 3.5 to 4.1 w/o U-235

R. S. Harding
R. J. Klotz
L. C. Noderer

Combustion Engineering, Inc.
Power Systems Group
Windsor, Connecticut

October 5, 1979

A. Summary and Conclusions

The amendment to SNM License 1067 justifying an increase in the maximum enrichment from 3.5 to 4.1 w/o U-235, which was submitted to the Nuclear Regulatory Commission under a cover letter dated April 23, 1979, and two subsequent transmittals of additional information dated June 27, 1979 and August 7, 1979 were reviewed from the standpoint of the technical acceptability of the criticality evaluations. The methods of review included detailed checking of input to selected analyses, checking of data sources and application of data to assessments of criticality safety by SIU methodology, and evaluation of acceptability of results by comparison with other analyses. The conclusions of this review are as follows:

1. On the basis of examining each of the analyses employing SIUs, 50% of the multigroup KENO calculations, and rationalization of the few group KENO calculations, it was concluded that the original criticality analyses shown in the amendment were carried out in a satisfactory way. No significant fault was found in any calculation.

2. Adequate subcritical margin exists in the various operations when handling 4.1 w/o enriched UO_2 providing all administrative controls and area limits are adhered to by Manufacturing personnel.

3. Detailed checking of selected Monte Carlo input resulted in only a few ambiguities which, when resolved, resulted in lower multiplication factors.

4. The use of 4 neutron energy groups in criticality evaluations tends to underestimate the multiplication factor for a specific low moderator density condition (0.05 gr/cc) by an amount which is dependent upon the geometry being analyzed. For the case of interest, in-plant storage of fuel assemblies, the calculated multiplication factors may be ~3% Δk nonconservative but the margin to criticality is calculated to be ~20% Δk.

B. Introduction

The Nuclear Manufacturing Facility in Windsor has prepared a license amendment to justify increasing the U-235 enrichment limit from 3.5 to 4.1 w/o U-235 in uranium dioxide. All steps in the fuel handling process were reviewed by Nuclear Manufacturing personnel, criticality evaluations for various operations were carried out under the direction of the supervisor of Nuclear Licensing and Safety, and a license amendment was prepared and transmitted to the U.S. Nuclear Regulatory Commission (NRC) under a cover letter dated April 23, 1979. In the cover letter it was stated that an independent review of all criticality safety calculations was being carried out under the direction of the C-E Nuclear Safety Committee and the results of this review would be forwarded under separate cover. The purpose of this document is to summarize the method and results of the review. The basic objective of this review was to determine whether the results of the criticality safety analysis were reasonable and acceptable.

The applicable standards, regulatory guides, or branch technical positions which are available for guidance in this review appear to be limited; consequently precedent and guidance from the NRC reviewer would appear to be a dominant factor in the overall review process. The only standard which appears to be pertinent is ANSI N 16.1-1975, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors. Section 4.2.5, Subcritical Limits, states that where applicable data are available, subcritical limits shall be established on bases derived from experiments, with adequate allowance for uncertainties in the data. Plots of limiting values of U-235 mass, cylinder diameter, slab thickness, volume, and areal density are given as a function of w/o U-235; these curves are based on the data compiled by H. K. Clark in Reference 1.

Safe masses and dimensions are defined therein as those values resulting in an effective multiplication factor which lies 0.02 below the average curve defined by analysis and normalized to experimental data. Deviations between individual experimental data points and average values of keff given by the curves fall within ±0.015 and for the most part within ±0.01, consequently one may deduce that an effective multiplication factor as high as 0.995 is acceptable under this standard. For this license application added conservatism is introduced in the definition of safe masses and dimensions beyond that defined by this standard.

Criteria on safe masses and dimensions are used quite extensively in this license amendment for demonstrating safe conditions for relatively simple geometries of fissile materials or for specific configurations which are either closely related to a configuration examined experimentally or convertible to a simple geometry through areal density techniques. Criticality evaluations of the more complex geometries are modelled on the computer using the Monte Carlo computer code, KENO IV.

Extensive compilations of experimental data are available for model verification (see for example, References 2, 3, and 4). However, there are noticeable gaps, the most prominent gap being for lattices moderated by low density hydrogeneous material so as to simulate so-called low density mist conditions. Mist conditions are postulated to occur for open arrays of fissionable material which may be accessible to, for example, fire fighting equipment such as sprinklers, foam or fog type nozzles. Typically, the most reactive moderator distribution is postulated so as to define the upper bounds to the multiplication factor. In reality, most fire fighting techniques are not able to give a sufficiently high density of mist so as to meet the postulated conditions. For example, overhead sprinklers can lead to conditions approximating water densities of 0.1% whereas fire fighting foams yield a hydrogen distribution equivalent to a water density of approximately 3% (special foam materials can go as high as 6%). Overhead sprinklers are employed in the manufacturing facility and fire hoses may be used but no foam is allowed for fire fighting in the facility. Consequently there is the likelihood that low density mist conditions could occur and, under certain conditions high local concentrations of water could occur for short periods of time, but the design and location of the facility is such that completeimmersion of manufacturing operations in an environment of water having an effective density of greater than 0.1 gr/cc is highly improbable.

The KENO analyses carried out in support of this license amendment are of two types: (1) 16 group calculations employing the Hansen-Roach cross-section libraries, and (2) 4 group analyses where the broad group cross-sections are derived by the CEPAK lattice code. Reference 5 has been cited as a basis for validating the use of the KENO with the Hansen-Roach cross-sections; however, the benchmark analyses were primarily reflected and unreflected plastic moderated criticals with uranium enrichments in the U-235 isotope of 5% or less and H/U-235 ratios in the range of ~130 to ~970. The absence of benchmark experiments on large geometry fissile arrays moderated by low density mist appears to be a persistent problem. The 4 neutron group approach employing CEPAK as the cross-section generator has been employed for criticality evaluations of spent fuel storage racks for many years. Appendix A provides information on the benchmarking of this technique. However, there are still valid questions as to the uncertainty of this 4 group technique when applied to fissile configurations containing a mist environment. This uncertainty is covered by allowing a large margin to calculated criticality.

Administrative controls play an important part in assuring that safe conditions exist in the fuel manufacturing facility. Since UO_2 fuel of less than ≤5 w/o U-235 cannot achieve criticality in the absence of hydrogeneous moderator and many of the operations in manufacturing are dry, a key objective of the administrative controls is to assure that in the event of flooding of the facility by water (so as to produce an environment comparable to part or full density water), fuel configurations are such that criticality safety is assured as demonstrated by the criticality safety analyses. Administrative controls and/or procedures include the following. All operations and changes in operations must be analyzed to establish safety limits and controls and these analyses must undergo independent review. Safety limits and controls are documented by written procedures. Signs defining the criticality safety limits are posted at each work station: in cases where the safe spacing area extends beyond the equipment boundaries, the boundary of the spacing areas are indicated by colored lines in the floor. All mass limited containers are labeled to indicate the enrichment and uranium content; procedures require labeling so that the identity of all the fuel enrichments is known throughout the facility. Line management, including the Nuclear Licensing and Safety supervisor, are responsible for enforcing all administrative controls.

C. Review Approach

1. Categorization of Criticality Evaluations

For purposes of reviewing the various criticality analyses, they are divided into four categories as listed in Table 1. The first category contains those analyses which use as their basis for demonstrating subcriticality the definition of Safe Individual Units (SIU). As noted earlier, safe masses and dimensions are defined by using experimental data to determine the magnitude of these parameters such that a subcritical multiplication factor is assured.

The second category consists of these criticality evaluations which employed the KENO-IV computer code and the 16 group Hansen-Roach cross-section library. The majority of these analyses involve homogeneous mixtures of fuel and moderator. The third category consists of those KENO evaluations which employed four neutron group libraries generated by the CEPAK code. These cases involve heterogeneous arrays of fuel, moderator, and in some cases, structural materials which were not amenable to volume homogenization. The fourth category involves those operations which are deduced to be subcritical by comparison with other evaluations.

2. Review Procedures

a. Safe Individual Units

The most expeditious way of reviewing those analyses employing SIU limits was determined to be an examination of the data sources and a check of each application under the criteria adopted for this license amendment.

b. 16 Group Monte Carlo Analyses

Four out of seven of the KENO analyses in this category were selected for detailed review of the analysis (items C3.3, C3.4, C6.1 and C6.2) on the basis that they postulated moisture in the fuel and involved the more complex geometries.

c. 4 Group Monte Carlo Analyses

Of the four analyses in this category, three exhibited maximum effective multiplication factors under full density water environments and the fourth, In-Plant Storage of Fuel Assemblies (C-8.7), exhibited its most reactive condition under a mist environment. The review of this category focused on the question of calculational uncertainties associated with the four group approach for mist conditions; in particular, what bias may result from the use of only 4 neutron groups rather than a larger number.

TABLE I Categorization of Criticality Evaluations

A. Safe Individual Units

19.1	Limits for Individual Units
19.2	Interaction Analysis
C-3.6	Pressing
C-3.7	Dewaxing and Sintering
C-3.8	Final Sizing
C-6.2	Pellet Storage Shelves - Additional Storage Evaluation
C-6.3	Transfer of Material
C-7.0	Pretreatment of Low Level Liquid Wastes
C-8.9	Fuel Salvage
C-8.10	In-Process Storage of Fuel Pellets in Containers
C-8.11	Rod Transfer

B. 16 Group Monte Carlo Analyses

C-3.2	Virgin Powder Storage Area
C-3.3	Batch Make-Up
C-3.4	Powder Preparation and Blending
C-6.1	Concrete Block Storage Area
C-6.2	Pellet Storage Shelves
C-8.4	Fuel Rod Storage Area
C-8.5	Double Shelf Rod Storage Racks

C. 4 Group Monte Carlo Analyses

C-8.1	Pellet Alignment and Drying
C-8.2	Rod Loading and Fuel Rod Transport Carts
C-8.7	In-Plant Storage of Fuel Assemblies
C-8.8	Shipping Container Storage

D. Comparative Analyses

C-3.5	Final Mixing
C-8.3	Autoclave Corrosion Test
C-8.6	Fuel Assembly Fabrication

D. Results of Review

1. Safe Individual Limits

A safe individual unit limit is that mass or dimension which characterizes either a heterogeneous or homogeneous array of fuel and moderator known to be subcritical (safe) by comparison with experimental data. The definition of safe masses and dimensions employed in this license application have added conse tism beyond that defined in ANSI-N-16.1. Section 19.1 quotes safety factors of 2.3, 1.3, 1.1, and 1.2 for mass, volume, cylinder diameter and slab thickness, respectively, as existing in the definition of safe individual unit limits. Attempts to verify the magnitude of the safety factors indicated that they may be underestimated relative to the data of Reference 2 but overestimated relative to the curves in ANSI-N-16.1. In any event it is concluded that there is substantial conservatism still remaining relative to either source of data. Additional conservatism has also been introduced in certain cases to meet criteria on maximum fraction critical values.

A review of section 19.2 concluded that Table 19.2 did not include the safe volume and spacing area for two story operation with 4.1 w/o UO_2 fuel to support, for example, the pressing operation in Exhibit C. Although the correct criteria are included in the text of 19.2 and the safe volume limit is given in Table 19.1, an additional statement in Table 19.2 is required for completeness. Based on a review of the references of section 19.1, it was concluded that the single level spacing criteria of Table 19.2 meet the fraction critical criteria outlined on page X1X-4. Comments on individual sections of Exhibit C employing the SIU approach are summarized below.

a. Section C-3.6 - Pressing

The writeup of this section of the license application is very brief and does not address criteria and controls for all aspects of the pressing operation. For example, interactions between hoppers and presses are not discussed. Actually the hoppers are designed to be safe cylinders (3.5 w/o) or safe volumes (4.1 w/o) and are located on a mezzanine above the presses. Criteria outlined in Sections 19.1 and 19.2 for application to two story operation indicate that the safe volume hopper for 4.1 w/o powder in combination with the already designated spacing areas for 3.5 w/o powder lead to safe conditions. The statements concerning maximum height of boats and a limit of only one boat at a press work station at any given time do not adequately describe the justification for safety of this operation. Actually the boats have a wall height of only 2 15/16 inches, the pellets are loaded to a point below the lip of the boat and a card is placed on top of the pellets giving pertinent data on the fuel. This procedure provides reasonable assurance that the safe slab limit of 3.7 inches is met. In addition, a boat loaded with pellets stacked to the height of the sidewalls of the boat contains approximately 12 kg UO_2 which is well below the safe mass limit of 24 kg for 4.1 w/o fuel.

b. Section C-3.7 - Dewaxing and Sintering

The criterion for safety of this operation is a safe slab limit of 3.7 inches which is achieved through controls discussed above on the height of pellets loaded in the boats.

c. Section C-3.8 - Final Sizing

Here again the safety criterion is a safe slab limit of 3.7 inches. In most steps of this operation the fuel is distributed in a layer having a thickness of ~1 pellet daimeter. The only part of the operation requiring investigation was the infeeder where the pellets are dumped into a flooded bowl, one boat at a time to meet the 3.7 inch safe slab limit. The volume of the infeeder bowl is 31.2 litres and for a volume ratio of H_2O/UO_2 less than 1.5, the critical volume is greater than 35 litres. Even if this bowl was fully filled with pellets it would be necessary to attain the large volume fraction of H_2O, consequently the bowl is a safe volume for this operation. At the output of the grinder, the pellets are placed into "grinder trays" having a height of 1.5 inches and a cover assembly. The paragraph on drying of grinder sludge does not state what control is employed; comments to the effect that the sludge is collected in volume limited SIU containers and trays having a maximum depth of 3.7 inches so that the oven is limited to a safe slab configuration, as stated in section 7.0, should have been included in the text. The centrifuge and grinder coolant sump meet the safe volume and spacing area criteria of sections 19.1 and 19.2 although the spacing area of the grinder coolant sump (9 ft^2) is not stated. For the storage rack (W.S. P-20), the safe slab limit is achieved by storing grinder trays no more than 2 high.

d. Section C-6.2 - Pellet Storage Shelves (W.S. P-15, 19, 21)

An increased slab thickness limit was adopted for this storage area so as to accommodate 3 levels of grinder storage trays. The consequence of using this increased slab thickness is reduced conservatism relative to section 19 criteria. The rationale for deducing the 5.5 inch slab limit was reviewed and found to be conservative relative to experiments by a factor of 1.20 on the slab thickness when the H_2O/UO_2 volume ratio is < 1.0. In reality, three levels of grinder storage trays would result in a fuel height of less than 4.5 inches which is significantly less than 5.5 inches but more than the 3.7 inch slab limit used in other operations.

e. Section C-6.3 - Transfer of Material

The SIU limit in combination with the cart dimensions meets the criteria in sections 19.1 and 19.2. Transfers by hand occur infrequently and are limited to one SIU. Attention is focused on cart transfers since carts may be left unattended and the dimension of the cart assures a safe spacing area. When material is transferred by hand it is generally either for a very short distance or consists of an amount of fuel less than contained in 1 SIU.

f. Section C-7.0 - Pretreatment of Low Level Liquid Wastes

The rationale employed to deduce the safety of this operation was reviewed and found to be acceptable. Although the diameter of the tank exceeds the safe cylinder limit (9.8" in Table 19.1) by 0.2", it is smaller than the critical diameter of the infinitely long, fully reflected cylinder having optimum moderation of the fuel water mixture by 0.8 inches. This margin in combination with the finite height of the settling portion of the tank (18 inches) should offer sufficient conservatism.

g. Section C-8.9 - Fuel Salvage

In addition to being mass limited, safe containers are employed to receive the recovered fuel.

h. Section C-8.11 - Rod Transfer

The rationale for the increase in the safe slab limit to 5.5 inches is discussed in paragraph 1d, above, on section C-6.2.

2. 16 Group Monte Carlo Analyses

The detailed review of the following analyses included checking the fuel composition for the most reactive case in each series of calculations for nuclide number densities, enrichment, potential scattering, dimensions, geometrical representations, and composition of other materials.

C 3.3 Batch Make-Up

C 3.4 Powder Preparation and Blending

C 6.1 Concrete Block Storage

C 6.2 Pellet Storage Shelves

The powder preparation and blending station involves a complex three dimensional geometrical representation in KENO which makes verification difficult. However, spotchecking of the geometry was done by reconstructing from the generalized geometry equations. The following comments are provided for the indicated sections.

a. Section C-3.2 - Virgin Powder Storage Area

The fact that interspersed water moderation and flooding were not addressed does not belong in the listing of conservative assumptions. The design of the facility itself provides the principal argument that external moderation is of no concern in this case. Internal moderation is addressed by the checking procedures of Section C-3.1 and the assumption in the criticality analysis of a higher value of moisture content than the limit defined in C-3.1.

b. Section C-3.3 - Batch Make-Up

The introduction of internal and external moderation as two separate variables raises questions as to the rate of convergence of the interative process and the validity of the process in general. Two iterations were carried out and there is no indication that the process is converging. However, it would appear from the trends of the analyses that any realistic conditions of moderation would result in much lower multiplication factors than computed here since: (1) the presence of near full density water external to the fuel containers is not probable under any conditions inside the hood, and (2) the optimum moderation conditions postulated for the fuel containers simultaneous with the postulated external moderation conditions are not realistic for this area of the facility.

c. Section C-3.4 - Powder Preparation and Blending

In subsection 3.4.2 under criticality analysis, the definition of optimum moderation should be stated more clearly as to the external moderation conditions. Once again the internal and external moderation variables approach is pursued. However, in this case the calculations appear to be convergent. The multiplication factors indicate a high degree of subcriticality even with extremely high degrees of moderation in both regions.

Drawing NFM-C-4065 shows 6 inches he separation of the two one-half inch thick fuel layers on conveyor belt whereas the calculation used 9 inches. The correct dimension is 9 inches; the drawing should be modified.

In the discussion of the front end of the station, clarification of the term "optimum moderation" as employed in the conservative assumptions would be beneficial. The iteration on internal and external moderation conditions was truncated after the first iteration on external moderator conditions and the resulting maximum multiplication factor is in the range of 0.94 to 0.95 for both enrichment cases. While there is less margin to criticality than in the cases discussed above, it is difficult to see how the flooding of these areas with full density water could occur.

In the review of the KENO geometry, two small wedges of fuel at the intersection of the powder spread funnel and the powder tubes appeared to be improperly described in the original definition of the material regions. This should have a negligible effect upon reactivity but the case has been rerun to assess the impact on the calculated multiplication factor. The revised multiplication factor was 0.8484 ± 0.0086 versus 0.8684 ± 0.0101 computed earlier.

d. Section C-6.1 - Concrete Block Storage Area

The review of the calculations uncovered no problems or questions other than those raised above pertaining to the treatment of internal and external moderation as separate variables.

e. Section C-6.2 - Pellet Storage Shelves

The discussion of the pellet storage shelves states that the shelves are limited to a slab thickness of 3.7 inches. Actually this is assured by limiting the number of the covered grinder trays stacked at any position to two which implies that the maximum fuel height is <3 inches. In the analysis, the iteration on internal and external moderation was truncated at the first iteration on external moderation and showed a relatively low maximum multiplication factor (~0.82) for an external moderation condition which is higher than one could attain from fire fighting equipment or a sprinkler system.

In the determination of cross-sections for the pellet-water mixture, a volume homogenization procedure was employed which is non-conservative for low enrichment fuel. However, it can be shown using data from Reference 2 that the non-conservatism of this approximation is more than offset by the assumption that the pellet water mixture is such that the fuel concentration is 2.7 gr U/cc. For a random loading of the trays, one would expect a higher fuel density, i.e. of the order of 5.9 gr U/cc. For 5 w/o UO_2 the following critical masses (kg) are deduced from Reference 2:

	gr U/cc	
	2.7	5.9
Homogeneous	~97	~570
0.4" dia. rods	~86	~230

From these data one can see that the conservatism associated with the assumption of a fuel density of 2.7 gr U/cc in the homogeneous approximation is equivalent to a reduction in the critical mass of 463 kg whereas the non-conservatism of using the homogeneous approximation evaluated for a fuel density of 5.9 gr U/cc is equivalent to only 340 kg.

The review of the KENO analyses indicated that the eight inch thick hollow concrete block wall was represented as full density concrete. The effect on reactivity is expected to be small but the case was rerun to evaluate the effect of reducing the wall thickness to five inches; the peak reactivity decreased from 0.8188 ± 0.0088 to 0.7791 ± 0.0081.

f. Section C-8.4 - Fuel Rod Storage Area

For UO_2 of 4.1 w/o enrichment and no hydrogeneous moderation, this area is clearly subcritical.

g. Section C-8.5 - Double Shelf Rod Storage Racks

This analysis is similar to that of section C-8.4 with the exception that the spacing between storage boxes is greater and no physical barriers have been used to exclude mist or personnel from between storage boxes. Previous analyses for the 3.5 w/o enriched fuel employed 4 neutron groups and the DOT code for the spatial calculation. For the case of flooded boxes the analyses yielded a maximum multiplication factor of ~0.87. In the case of 4.1 w/o fuel but with no water inside the boxes, a multiplication factor of ~0.89 was obtained. These two sets of results are in reasonable agreement if one assumes that the non-conservatism inherent to the use of only 4 neutron groups with mist between boxes is nearly offset by the increase in enrichment from 3.5 to 4.1 w/o and the elimination of the relatively small amount of room temperature water inside the boxes.

3. 4-Group Monte Carlo Analyses

a. Validity of 4 Neutron Group Model

The analytical model employed in the criticality evaluation of fuel manufacturing operations considered in this section used CEPAK to generate 4 neutron group cross-sections and KENO to solve for the spatial solution of the multiplication factor. Validation of this model is discussed in Section 3.0 of Exhibit D of the license application. Results discussed in Exhibit D indicated very good agreement for lattices employing full density water. Analyses discussed in Appendix A of this report provide added assurance of the validity of basic cross-section libraries and general methods of analyzing the reactivity of a broad variety of lattices. Thus, there is a high degree of confidence in the methodology not only at the full density water conditions but also at the reduced density conditions existing in a hot, full power reactor environment, i.e. water densities down to ~0.70.

A question arises as to the accuracy of the 4 group approach for mist conditions, viz. how strong is the dependence of the multiplication on the number of broad neutron groups employed when two closely interacting regions of differing neutron spectrum are involved. To examine this point, analyses of a 14x14 fuel assembly in a mist environment (0.05 gr/cc of H_2O) were carried out using the DOT code for the spatial solution and alternate cross-section generators to prepare broad group cross-sections in differing numbers of neutron groups. It should be noted that the CEPAK lattice code is not employed to generate broad group cross-sections in more than 3 non-thermal groups since it employs the MUFT type solution to solve the multigroup equations. The GAM code was used as an alternate method of deriving the non-thermal broad group constants and, in one lattice geometry, the DIT code was employed to generate group constants. Both the GAM-THERMOS and DIT calculational models employed the same basic multigroup neutron cross section libraries (ENDF/B-IV); however, differences in resonance self-shielding did exist between the two models.

The DIT code is a C-E proprietary code which solves for the multigroup neutron spectrum in 85 neutron groups throughout an entire fuel assembly and not simply the fuel pin cell as with CEPAK or GAM. It is presently employed for reactor lattice calculations where it solves for the spatially dependent multigroup spectrum in various subregions of the heterogeneous fuel assembly and provides few group cross-sections for specified subregions. The DIT code is not presently programmed to deal explicitly with the large geometrical arrays encountered in the present calculations, but it can be used to generate few group cross-sections more representative of spectral variations than either CEPAK or GAM-THERMOS. This capability was exploited in the generation of 4 and 9 group cross-sections for a DOT calculation of an infinite array of 14x14 fuel assemblies having an edge-to-edge spacing of 12 inches and a 0.05 gr/cc mist both internal and external to the fuel assembly.

Figure 1 shows a comparison of the group dependence of the multiplication factor computed with DIT and GAM-THERMOS derived cross-sections in the DOT code. To expedite the GAM-THERMOS calculations, a modified 14x14 cell definition was used to represent the fuel assembly which preserved the volume of structure, fuel and moderator but redistributed the moderator and structure from the CEA guide tube regions to the moderator region of the unit cells. CEPAK and GAM were written to use slightly different resonance self-shielding algorithms; the former employs a fit to a Hellstrand experimental correlation whereas the latter used the Nordheim Integral Treatment based on resolved and unresolved resonance parameters with a CEPAK derived Dancoff factor. The resulting difference in multiplication factors can be seen by comparing the 4 group "CEPAK(HETEROG)" point with the curve labeled GAM-THERMOS(HOMOG). To show the few group trend, the latter curve was displaced downward to pass through the CEPAK point and is plotted as a broken line. The points of interest in Figure 1 are as follows. First, the DIT calculation shows very little "few group" dependence because the multigroup solution for the spatially dependent neutron spectra includes the influence of the mist environment in the generation of the few group constants. Second, the classical Wigner-Seitz cell approximation for the fuel pin and a separate calculation of reflector constants employed in the GAM-THERMOS approach does not properly account for the fuel-reflector interaction in the generation of few group constants under the assumed mist conditions. Increasing the number of neutron groups gives a better approximation to the energy dependent spatial flux solution but may still underestimate the multiplication factor because of a failure to adequately represent the effect of the interassembly mist environment into the calculation of primarily the resonance escape calculation.

Figure 2 shows the results of the DOT calculations for few group cross-sections derived by both GAM-THERMOS and CEPAK for the larger interassembly spacings employed in the storage of fuel assemblies in the manufacturing plant; no DIT based calculation could be carried out for this configuration since the problem size exceeds present capabilities. The first point of interest is that the group dependence is much less than in the closer spaced geometry of Figure 1. The second point is that the bias in the calculation due to using only 4 neutron groups appears to be in the range of 3% Δk.

Analyses at full density water conditions exhibit significantly less dependence on the number of "few groups" employed.

b. Comments on Applicable Sections

(1) Section C-8.1 - Pellet Alignment and Drying

The first paragraph states that the pellet configuration is limited to a 3.7 inch slab thickness; the text could state that this is an administrative control and the layer of pellets on the table is generally one pellet high.

Item 5 of the assumptions stated that the aluminum pellet troughs were not included in the analysis; no reference was made about the remaining aluminum structure inside the ¼ inch thick stainless steel cylinder. The mean spacing of the pellet columns is sufficiently large that the assumed representation of the fuel/mist lattice is over-moderated in the fully flooded case. Therefore, whether or not substitution of moderator for aluminum structure is a conservative assumption may depend upon the actual configuration of fuel, structure and moderator. In view of the latter question and the absence of data points between water densities of 0.5 and 1.0 in Figure D-1.7.1, a review was made of the analyses carried out to develop Figure D-1.9.3 (Rev. 2, 9/16/74) which is labelled "Critical and Safe Cylinder Diameters as a Function of Rod Spacing* and Moderator Density". Although these analyses are for 3.5 w/o fuel rods, they are informative. They indicate that indeed the oven, if fully loaded and flooded with full density water, is overmoderated and more than 500 rods at the average spacing employed in the oven would be required for a critical configuration. The difference in reactivity due to the uranium enrichment from 3.5 to 4.1 w/o should be more than offset by the introduction of the poison rods. Consequently, if the oven could be fully flooded, it should be subcritical.

Item 6 of the conservative assumptions states that four group cross sections were generated by the CEPAK code for fuel and poison regions; actually a HAMMER-DTF sequence of calculations was used for these lumped poisons. The discussion at the top of page 19b is unclear as to whether the internal and external moderator conditions were varied simultaneously or independently; actually they were treated as a single variable.

(2) Section C-8.2 - Rod Loading and Fuel Rod Transport Carts

The description of the fuel configuration is not too clear; actually there are 250 fuel rods arranged in 5 concentric rings formed by 4 spacer rings within an annulus of ~14.6" ID and 25.7" OD. The analytical results are judged to be reasonable on the basis of the relative large mean spacing of fuel rods, the maximum number of rods (250) and decoupled (annular) array of fuel rods in the fixture.

*caption on figure says rod size other than rod spacing.

(3) Section C-8.7 - In-Plant Storage of Fuel Assemblies

The postulated representation of the fuel array for this category of analyses resulted in a maximum multiplication factor for reduced water density conditions ($\rho < 0.5$ gr/cc), of ~0.80. Consequently a margin of approximately 0.20 Δk exists to the critical condition. This margin should be more than adequate to cover the bias associated with the use of 4 neutron groups as well as other biases and calculational uncertainties.

(4) Section C-8.8 - Shipping Container Storage

If the shipping containers are placed in contact there will be a minimum of approximately 14 inches separating fuel assemblies in adjacent containers. Under fully flooded conditions this is adequate to prevent interaction. Based on analyses of fuel transfer tubes and upenders, the calculated multiplication factor is reasonable.

4. Comparative Analyses

a. Section C-3.5 - Final Mixing

The writeup neglects to state that the spacing area for this operation is 27 ft^2 which meets the requirement of Section 19 of the License.

b. Section C-8.3 - Autoclave Corrosion Test

The rationale for criticality safety contained in the license application is in error on one point, viz. that the fuel rods in a fuel assembly are spaced at the most reactive pitch. Actually engineering judgement would say that the autoclave is highly subcritical with 32 fuel rods present. The previous defense of this operation for 3.5 w/o fuel showed that the system was subcritical with 120 fuel rods. For 4.1 w/o fuel the number of rods has been reduced by 73% and the amount of U-235 has been increased by only 17%; in addition, the rod spacing has been increased so as to further decrease the infinite multiplication factor below that which would exist if the rods were spaced so as to accommodate 120 rods in the autoclave. Clearly this system will be subcritical.

c. Section C-8.6 - Fuel Assembly Fabrication

The logic employed for deducing safety of this operation is acceptable.

References

1. H. K. Clark, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," DP-1014, Savannah River Laboratory (1966).

2. UKAEA Handbook AHSB(1) - Handbook of Criticality Data, Compiled by J. H. Chalmers, G. Walker, and J. Pugh, United Kingdom Atomic Energy Authority (1965).

3. G. A. Price, "Uranium-Water Lattice Compilation Part I, BNL Exponential Assemblies," BNL-50035, Brookhaven National Laboratory (1966).

4. E. Bernocchi and R. Martinelli, "Light Water Lattice Data," NEACRP-U-190, CNEN-Quaderno KIT/FIS (77)1.

5. G. R. Handley and C. M. Hopper, "Validation of the "KENO" Code for Nuclear Criticality Safety Calculations of Moderated, Low-Enriched Uranium Systems," Y-1948, Oak Ridge Y-12 Plant (1974).

Figure 1 - Multiplication Factor vs. Number of Neutron Groups for 12" Edge-To-Edge Spacing of 14X14 FA; 0.05 gr/cc Mist Moderation.

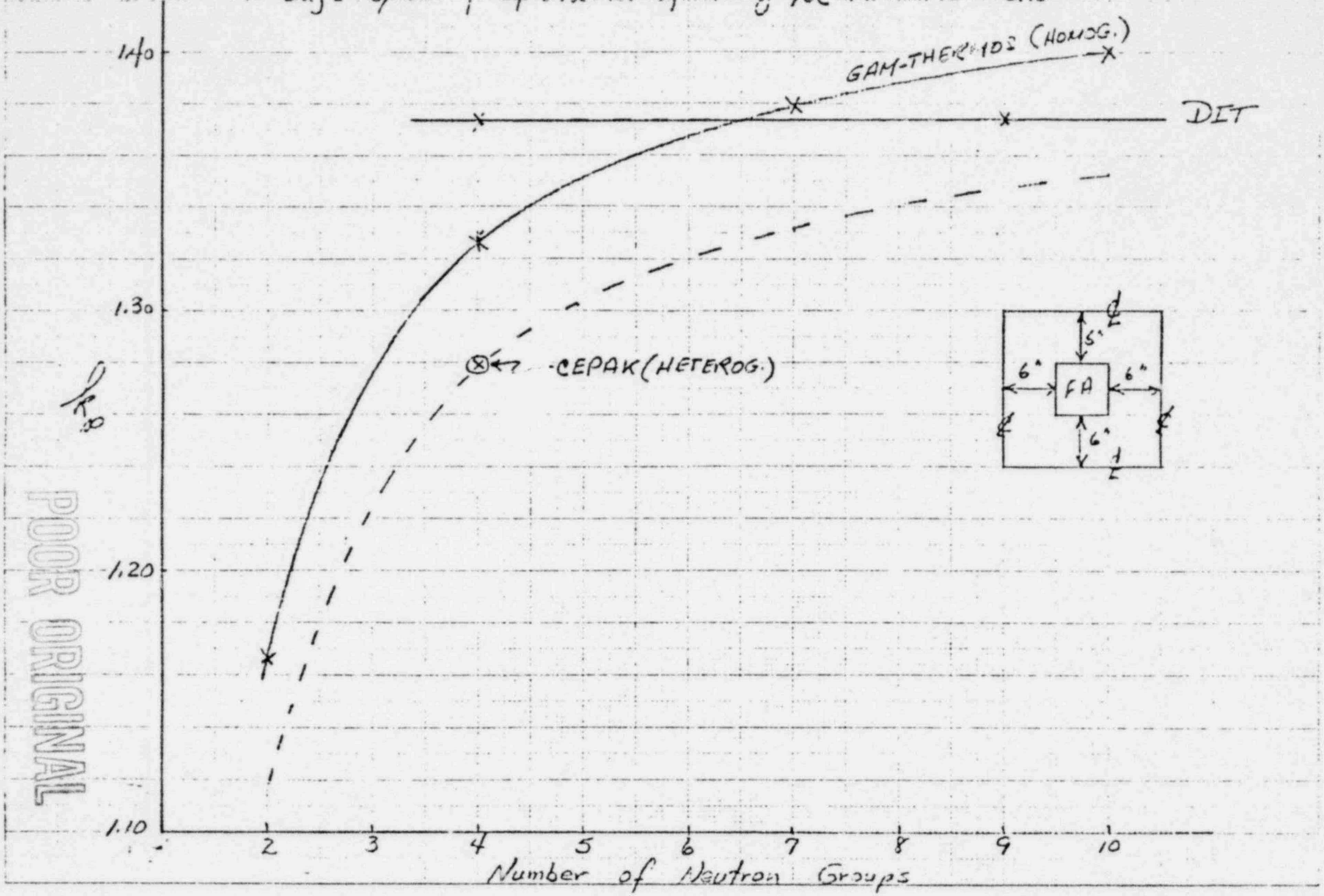
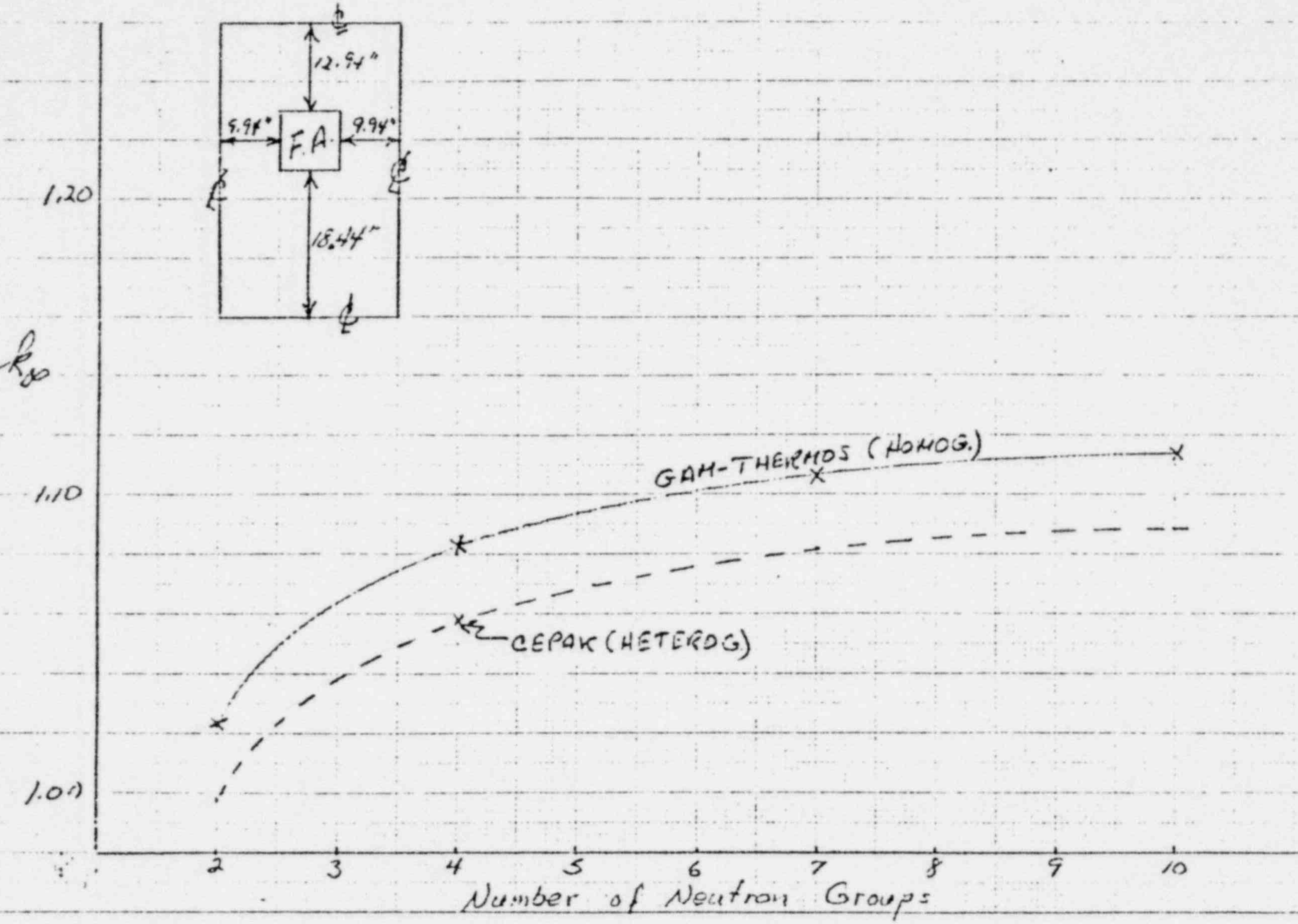


Figure 2 - Multiplication Factor vs. Number of Neutron Groups for F.A. Spacing Typical of In-Plant Storage of Fuel Assemblies; 0.05 gr/cc Mist Moderation



APPENDIX A

Qualification of Analytical Method Employed in Criticality Evaluation of Fuel Handling Operations

I. Purpose

The purpose of this Appendix is to provide qualification of the calculational model and evaluation of calculational uncertainties and/or bias factors used in performing criticality evaluations of fuel handling operations with structural and/or fixed poisons in the form of steel boxes and boron carbide plate. This qualification is based on the analysis of a variety of reactor and laboratory experiments. The methods of cross section generation are essentially those of C-E's physics design procedures modified appropriately for use in four group transport, discrete ordinate method criticality calculations, and Monte Carlo codes.

II. Calculational Uncertainty and Bias

The results of the analysis of a series of UO_2 critical experiments are summarized in Table I. These are calculated using the CEPAK 2.3 lattice code as a few group neutron cross-section generator. Table I includes the mean and standard deviation for this CEPAK model. These calculations support use of the differential cross-section data base and broad group cross-section generation codes.

To assess the accuracy of the calculational model in predicting the multiplication factor of fuel assemblies having a separation distance sufficiently large so as to be isolated, analyses were carried out for a group of subcritical exponential experiments on clusters of 3.0 w/o UO_2 fuel pins clad with type 304 S.S. and moderated by H_2O (page 165 of Reference 7). The cluster sizes analyzed vary from 181 to 301 fuel rods so as to encompass the range of sizes typical of current PWR fuel assemblies. In these analyses, the spatial flux solution was obtained directly with the transport code, ANISN. The multiplication factors for the lattices analyzed using axial bucklings deduced from the reported relaxation lengths are tabulated below.

No. of Fuel Rods	K_{eff}
181	0.9966
211	1.0011
235	0.9966
265	0.9983
301	0.9984

These results indicate that the calculational model predicts the multiplication factor for small clusters of fuel rods in a water environment to a high degree of accuracy, i.e. a bias of -.0017.

To ascertain whether the calculational model can predict the reactivity characteristics of subcritical clusters of fuel separated by water channels of various thicknesses and, in some cases, with thick stainless steel plates and boron poisoned plates inserted in the water channels, an analysis was made of the experiments on critical separations of 2.35 w/o U-235 UO_2 subcritical clusters reported in Reference 8. The results using the Monte Carlo code KENO IV are shown in Table II.

The calculation methods for these experimental comparisons, which are also used in criticality evaluations for fuel storage racks, fuel shipping containers, plus other fuel configurations found in fuel manufacturing areas, are based on CEPAK 2.3 cross-sections. Using an appropriate buckling value and taking proper account of resonance absorption, three fast groups are collapsed from 55 fine energy mesh groups in FORM and the one thermal group is collapsed from 29 thermal energy groups in THERMOS. In addition, each component such as water gap, or poison plate has its thermal cross-section determined by a slab THERMOS calculation employing an appropriate fuel environment. FORM and THERMOS are sub-programs of CEPAK.

For one dimensional analyses such as the BNL exponential experiments the discrete ordinates code ANISN (Reference 9) is used. For two dimensional analyses DOT-2W (Reference 10) is used. For three dimensional analyses (such as the critical separation experiments) KENO IV (Reference 11) is used.

The abov analyses indicate a mean error between predicted and measured multiplication factors of +.00135 and a calculational uncertainty of 0.00714 at the 95/95 confidence level for the complete series of UO_2 experiments.

References:

1. T. C. Engelder, et al, "Spectral Shift Control Reactor, Basic Physics Program," B&W-1273, November 1963.

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

3. P. W. Davison, et al, "Yankee Critical Experiments," YAEC-94, April, 1959.

4. W. J. Eich and W. P. Rocacik, "Reactivity and Neutron Flux Studies in Multi-Region Loaded Cores," WCAP-1443, 1961.

5. F. J. Fayers, et al, "An Evaluation of Some Uncertainties in the Comparison Between Theory and Experiments for Regular Light Water Lattices, Brit. Nuc. En. Soc. J., 6, April 1967.

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7. G. A. Price, "Uranium - Water Lattice Compilation Part I, BNL Exponential Assemblies," BNL-50035 (T-449), December 1966.

8. S. R. Bierman, E. D. Clayton and B. M. Durst, "Critical Separation Between Subcritical Clusters of 2.35 w/o U-235 Enriched UO_2 Rods in Water With Fixed Neutron Poisons," PNL-2438, October 1977.

9. Ward W. Engle, Jr., "A Users Manual for ANISN, A One Dimensional Discrete Ordinates Transport Code With Anisotropic Scattering K-1693, March 30, 1967.

10. R. G. Sottesy, R. K. Disney, A Collier, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code," WANL-TME-1982, December 1969.

11. L. M. Petrie and N. F. Cross, "KENO IV, An Improved Monte Carlo Criticality Program," ORNL-4938, November 1975.

TABLE I

Results of Analysis of Critical UO_2 Systems

No.	Lattice		B^2_{tot}	K_{eff}*
1	B&W (1)	I	.88-2	1.00121
2		II	.172-2	1.00534
3		X	.79-2	.99838
4		XIII	.701-2	1.00419
5		XX	.202-2	1.00550
6	B&W (2)	1	.861-2	1.00269
7		2	.420-2	1.00443
8	Yankee (3)	1	.408-2	1.00088
9		2	.531-2	1.00115
10		3	.633-2	1.00136
11	Yankee (4)	4	.688-2	1.00244
	Winfrith (5)			
12		R1-20	.660-2	1.00214
13		R1-80	.626-2	.99942
14		R3	.510-2	1.00422
15	Bettis (6)	1	.326-2	1.00053
16		2	.355-2	1.00046
17		3	.342-2	1.00106
	Average			1.00208
				±.00206

* Using calculated radial bucklings and measured axial bucklings.

TABLE II

Calculated k_{eff} Values
For Separation Experiments

Expt #	Type Poison Plate	K_{eff}	Monte Carlo σ (STD Deviation)
15	None	1.00227	.00534
04	None	0.99912	.00540
49	None	1.00221	.00473
18	None	1.00813	.00489
21	None	0.99589	.00461
28	304 S Steel 0.0 w/o Boron	1.00393	.00308
05*	304 S Steel 0.0 w/o Boron	1.00329	.00303
29	304 S Steel 0.0 w/o Boron	1.00271	.00302
27	304 S Steel 0.0 w/o Boron	1.00418	.00273
26	304 S Steel 0.0 w/o Boron	0.99811	.00279
34	304 S Steel 0.0 w/o Boron	0.99793	.00297
35	304 S Steel 0.0 w/o Boron	1.00436	.00290
32	304 S Steel 1.05 w/o Boron	0.99970	.00524
33	304 S Steel 1.05 w/o Boron	1.01173	.00491
38	304 S Steel 1.62 w/o Boron	1.00289	.00512
39	304 S Steel 1.62 w/o Boron	1.00208	.00506
20	Boral	0.99585	.00301
16	Boral	1.00020	.00288
17	Boral	0.99519	.00286

Mean K_{eff} Value 1.00157

Std. deviation .00419