Entergy

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1CAN110302

November 21, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement to Amendment Request
To Changes for the Spent Fuel Pool Loading Restrictions
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES: 1. Entergy letter to the NRC dated April 2, 2003, License Amendment Request to Modify the Fuel Assembly Enrichment, the Spent Fuel Pool (SFP) Boron Concentration Technical Specification (TS) 3.7.14, the Loading Restrictions in the SFP in TS 3.7.15, and to Modify the Fuel Storage Design Features in TS 4.3 (1CAN040302)

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs) to increase the fuel assembly enrichment and to modify the spent fuel pool to allow the insertion of Metamic poison panels.

On June 11, July 3, July 22, and August 7, 2003 Entergy received requests for additional information (RAI) from the Reactor Systems Branch, the Materials and Chemical Engineering Branch, the Mechanical and Civil Engineering Branch, and the Plant Systems Branch, of the NRC, respectively. Entergy's responses to these questions are contained in Attachments 1, 2, 3, and 4 to this letter.

Due to manufacturing difficulties, the original design of the support structure for the Metamic panels was altered to add additional stainless steel into the frame of the Metamic insert. The final design of the inserts resulted in minor changes to the reactivity analysis, which is reflected in an update of Chapter 4 of the licensing report and is included as Enclosure 1 to this letter. Changes to Chapter 4 are marked with revision bars. There were no changes to Appendix 4A, Benchmark Calculations, which is referenced in Chapter 4; therefore, it is not included in this letter. The design change also modified the structural analysis, which resulted in the need to provide a new structural analysis. The new structural analysis will be provided at a later date and will supercede the analysis that was contained in the original submittal (Reference 1). There was no impact on the other chapters of the Holtec Licensing Report.

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There are no technical changes proposed. The original no significant hazards consideration included in Reference 1 is not affected by any information contained in this supplemental letter. There are new commitments contained in this letter.

If you have any questions or require additional information, please contact Dana Millar at 601-368-5445.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 21, 2003.

Sincerely,

Sherrie R. Cotton

Sherrie R. Cotton

SRC/dm

Attachments:
1. Response to Reactor Systems Branch Request for Additional Information
2. Response to Materials and Chemical Engineering Branch Request for Additional Information
3. Response to Plant Systems Branch Request for Additional Information
4. Response to Mechanical and Civil Engineering Branch Request for Additional Information
5. List of Regulatory Commitments

Enclosure 1 – Holtec Report HI-2022867 Chapter 4 Revised

cc: Dr. Bruce S. Mallett
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U. S. Nuclear Regulatory Commission
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Arkansas Department of Health
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Attachment 1

To

1CAN110302

Response to Reactor Systems Branch
Request for Additional Information

Response to Reactor Systems Branch Request for Additional Information Related to Changes to the Spent Fuel Pool Loading Restrictions

Question 1:

The licensee's amendment described a methodology used to calculate the maximum effective multiplication factor (k_{eff}). The staff has outlined two acceptable methodologies to perform spent fuel pool criticality analyses in a letter entitled "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," from L. Kopp to T. Collins dated August 19, 1998. The two methodologies are the following: 1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or 2) a sensitivity study of the reactivity effects of the tolerance variations. The licensee's amendment did not clearly identify which methodology was used. The staff requests the licensee identify which methodology was employed to calculate the maximum k_{eff}.

Response 1:

The spent fuel pool (SFP) criticality analysis was performed using the second methodology; a sensitivity study of the reactivity effects of the tolerance variations.

Question 2:

The licensee calculated maximum effective multiplication factors by statistically combining all of the reactivity effects due to tolerances and uncertainties for each of the Arkansas Nuclear One, Unit 1 (ANO-1) spent fuel pool regions. However, the licensee's amendment does not contain the equations used to calculate these values. The staff requests the licensee provide the equations used to perform the maximum k_{eff} calculations and a detailed quantitative example demonstrating how the reactivity effects of each tolerance and uncertainty was calculated. The licensee's example should clearly and numerically demonstrate the methodology used to calculate the reactivity associated with each uncertainty or tolerance. Additionally, the staff requests the licensee calculate the values presented in one of the reference cases of the amendment as the example. The licensee should include a detailed description of the statistical methods employed and the values used in the calculation of any statistical uncertainties.

Response 2:

An equation could be written as:

$$\text{Maximum } k_{eff} = k_{eff}\ (\text{calculated}) + \text{biases} + [\Sigma i\ (UNC_i)^2]^{1/2}$$

where "UNC" values are the reactivity effects of individual uncertainties, each evaluated at the 95% probability, 95% confidence level or greater. (See also the ANSI/ANS 8.17 standard.)

A detailed example of the calculation of tolerance effects is shown in attached Table R-2.1 (listing the major tolerance values) and in Table R-2.2 (reactivity effects of each individual tolerance). Each of the reactivity effects shown in Table R-2.2 were calculated, one at a time, with CASMO4 by setting each tolerance at its most reactive value in a parametric study as indicated in these tables. Table 4.2.1 of the licensing report (reproduced here as Table R-2.3) shows how the manufacturing tolerance uncertainties are statistically combined with other uncertainties and added to the k_{eff} and biases from the basic calculation. Every other region in the SFP is treated in a similar methodology to obtain the reported uncertainties and maximum reactivity values. All of the individual reactivity uncertainties are combined using the standard conventional statistical methodology, i.e., square root of the sum of the squares of the independent uncertainties. Since each of the reactivity uncertainties are independently calculated at the 95% probability/ 95% confidence level or greater, the final maximum k_{eff} is at the 95% probability/ 95% confidence level, in accordance with the specifications of the 1978 Grimes letter and the 1998 Kopp memorandum.

Table R-2.1 Manufacturing Tolerances

Data	Tolerance Value
Box I.D.	See Figures 4.3.1 and 4.3.2, of Report HI-2022867
Box Wall Thickness	±0.004 inches
SS Thickness	±0.004 inches
Water Gap for Region 1	±0.01 inches
Metamic Width	±0.0625 inches
B-10 Loading	±0.5% B_4C
Metamic Thickness	±0.003 inches

Table R-2.2 Reactivity Effects of Manufacturing Tolerances in ANO- 1 Region 1 Spent Fuel Racks for 4.95% Enrichment Fuel Assemblies at 0 ppm Soluble Boron and 0 Years Cooling Time in the SFP.

Burnup, GWD/MtU	Reference	Minimum Box Id		Min. Water Gap		Min. Box Wall		Fuel Density		Min Sheath		Statistical Sum
	k_{inf}	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	
0	1.2129	1.2135	0.0006	1.2132	0.0003	1.2156	0.0027	1.2142	0.0013	1.2160	0.0031	0.0044
10	1.1318	1.1324	0.0006	1.1320	0.0002	1.1343	0.0025	1.1327	0.0009	1.1347	0.0029	0.0040
20	1.0686	1.0692	0.0006	1.0689	0.0003	1.0711	0.0025	1.0697	0.0011	1.0714	0.0028	0.0040
30	1.0097	1.0102	0.0005	1.0099	0.0002	1.0120	0.0023	1.0112	0.0015	1.0124	0.0027	0.0039
40	0.9529	0.9534	0.0005	0.9532	0.0003	0.9552	0.0023	0.9550	0.0021	0.9555	0.0026	0.0041
50	0.8987	0.8991	0.0004	0.8989	0.0002	0.9008	0.0021	0.9015	0.0028	0.9011	0.0023	0.0042

Table R-2.3

Summary of the Criticality Safety Analyses for Storage of Spent Fuel Assemblies in ANO-1 Region 1 Racks

Reference k_{eff}	0.9710
Burnup, MWD/KgU	35.4
MCNP4a Bias	0.0009
Temperature Bias	0.0100
Uncertainty in MCNP4a Bias	±0.0011
MCNP4a Statistical (95/95) Uncertainty	±0.0007
Manufacturing Tolerance Uncertainty	±0.0041
Enrichment Tolerance Uncertainty	±0.0027
Depletion Uncertainty	±0.0119
Fuel Eccentric Positioning Uncertainty	±0.0020
Statistical Combination of Uncertainties	±0.0131
Maximum k_{eff}	0.9950
Regulatory Limiting k_{eff}	1.0000

Question 3:

The licensee's amendment included a mechanical tolerance uncertainty term in the calculation of the maximum k_{eff}. However, the specific details of which uncertainties were included and how they affected the criticality analysis were not provided. The staff requests the licensee provide a table containing the following: 1) the mechanical tolerances considered, 2) the value for the tolerance used in the analysis, 3) whether the tolerance represents a maximum/minimum acceptable value or a statistical uncertainty, and 4) the resulting change in reactivity which can be attributed to the tolerance.

Response 3:

The response to question 2 presented an example of the mechanical tolerances considered, their values, their effect on reactivity, and the statistical combination of uncertainties, biases, and the reference calculation to define the maximum reactivity. Other cases are described in detail in the following tables given in the report supporting the licensing amendment.

Table 4.2.1	Summary of the Criticality Safety Analyses for Storage of Spent Fuel Assemblies in Region 1 Racks
Table 4.2.4	Summary of the Criticality Safety Analyses for 2-of-4 Checkerboard Storage of Fresh Fuel Assemblies and Empty Cells in Region 1 Racks
Table 4.2.5	Summary of the Criticality Safety Analyses for Storage of Spent Fuel Assemblies in Region 2 Racks
Table 4.2.8	Summary of the Criticality Safety Analyses for 2-of-4 Checkerboard Storage of Fresh Fuel Assemblies and Empty Cells in Region 2 Racks
Table 4.2.9	Summary of the Criticality Safety Analyses for Storage of Fresh Fuel Assemblies in ANO Unit 1 Region 3 Racks

These tables document details of the reactivity effects of the mechanical tolerances, and how they are combined with other tolerances (such as the enrichment tolerance) and added to the bias and k_{eff} from the reference calculation to obtain the maximum reactivity (k_{eff}).

All tolerances are inherently statistical in nature (See also NBS (now NIST) Handbook 91, *Experimental Statistics*, by M.B. Natrella). When there is a sufficiently large population of values (as in MCNP or KENO calculations) it is possible to determine the uncertainty at the 95% probability/ 95% confidence level. In some cases, the absolute maximum/minimum tolerances that exceed and conservatively bound the 95%/ 95% value may be used (for example, the ASTM tolerance limits for stainless steel plate thickness, ASTM SA 480/SA-480M). The tables cited above show the change in reactivity due to tolerances that might possibly increase or decrease reactivity on a statistical basis.

Question 4:

The licensee's amendment identifies Babcock and Wilcox (B&W) 15 x 15 spent and fresh fuel assemblies as the fuel type stored in the spent fuel pool and, therefore, used in the criticality analysis. The staff requests the licensee specify whether any other fuel types are currently stored in the ANO-1 spent fuel pool. If additional fuel types are stored in the pool, the staff requests the licensee demonstrate quantitatively that the B&W 15x15 assemblies provide the most conservative criticality analyses. Additionally, if B&W 15x15 assemblies are currently the only type of fuel stored in the spent fuel pool, the staff requests the licensee describe what licensing actions will be taken to amend the spent fuel pool licensing basis if different fuel types are used in the future.

Response 4:

No fuel types other than B&W 15 x 15 fuel assemblies are currently stored in the ANO-1 SFP. If different fuel types are used in the future, changes to the fuel assembly design, key fuel assembly mechanical features, and the changes in operating strategy will be evaluated under 10 CFR 50.59, "Changes, tests and experiments."

Question 5:

The licensee credits fuel burnup and cooling time to permit storage of higher enrichment spent fuel assemblies within the pool. The licensee's amendment does not state whether interpolation between the curves of burnup and cooling time versus enrichment will be used. The process of interpolating between curves on a graph introduces additional error not accounted for in the licensee's analysis. If the licensee intends to interpolate between the curves, the staff requests the licensee identify the maximum uncertainty possible from the interpolation and account for it with an appropriately conservative reactivity addition to the criticality analyses.

Response 5:

The curves presented in the Holtec License report are a pictorial presentation of the data, intended to aid in understanding the restrictions and the effect of important variables. Reading a graphical representation cannot be used for more than an approximation in determining the acceptability of a fuel assembly for storage in the racks. If an assembly exceeds the limiting burnup or cooling time by a large margin, it would be clear and obvious that the assembly is acceptable for storage. The great majority of fuel assemblies in storage fall into this category. If the burnup and/or cooling time of a particular assembly is close to the limit, a more precise and conservative method of interpolation is necessary, as provided by the polynomial fits which are reflected in Table 4.2.3 (Region 1) and Table 4.2.7 (Region 2) of the Holtec License report. These polynomial fits provide a conservative determination of the required fuel burnups that always bound the limiting values and do not require additional interpolation uncertainty. The proposed change to the Technical Specification (TS) bases includes the formulas for the decay time curves. If a fuel assembly's burnup and/or cooling time are close to the limit, the formulas provided in the TS bases would be used to ensure an accurate interpolation.

Question 6:

The licensee's amendment request stated that conservative values of moderator and fuel temperature and average soluble boron concentration were used in its burnup calculations. The staff requests the licensee provide a table listing the values used in the analysis and a technical justification for why these will generate conservative burnup values.

Response 6:

The parameters used in the depletion analyses are listed in Table R-6.1 below. These values were taken as the upper bound (most conservative) of the core operating parameters. The neutron spectrum is hardened by each parameter and hence the greatest production of plutonium during burnup, which results in the most conservative reactivity values.

Table R-6.1

Input Parameters Used in Depletion Calculations

Parameters	Values
Average T_{fuel}	1010 °F
Moderator $T_{average}$	578.93 °F
Moderator T_{outlet}	604 °F
Cycle Average Soluble Boron Used	900 ppm

Question 7:

The licensee has placed considerable emphasis on credit for burnup of the spent fuel for storage in the Region 1 and 2 racks. The staff requests the licensee provide detailed information describing the methods that will be in place, either administratively or experimentally, to independently confirm the fuel burnup before an assembly is placed in the storage racks.

Response 7:

ANO administratively controls the proper storage of fuel assemblies within the spent fuel pool racks and the classification of fuel assemblies based upon their accumulated burnup. The bumup values for each assembly are calculated by the plant core monitoring program. This quality related code calculates burnup values based upon incore detector readings. A qualified engineer takes the burnup values and reduces the burnup value to account for the instrument uncertainties. If an alternate source of burnup data is used, for example the use of burnup measurement equipment, then the uncertainty associated with determining the burnup will be determined. The adjusted burnup value is then used in conjunction with the fuel assembly enrichment and discharge time to calculate the fuel assembly Region 1 and Region 2 classifications based of the decay time curves in the proposed Technical Specification. This entire process for classifying fuel assemblies is independently verified by a qualified engineer.

Question 8:

The licensee's analysis stated that Region 3, which will contain the Metamic® inserts, had a negative moderator temperature coefficient (MTC), while the analyses for Region 1 and Region 2 had a positive MTC. In the Region 3 analysis, the licensee treated temperatures *less than 20 °C as uncertainties and statistically combined its reactivity effect with other uncertainties in the criticality analysis, while in the Region 1 and 2 analyses higher temperatures were treated as biases and directly added to the calculated k_{eff} value. Temperature was not a nominal design value nor a tolerance limit in this analysis; instead, it was a reference point for the calculation using the MCNP4a code. The differences in how the licensee includes reactivity variations due to temperature differences results in an inconsistent licensing basis and a non-conservative maximum k_{eff} in the Region 3 criticality analysis. Therefore, it is inappropriate to handle temperatures less than 20 °C as uncertainties, while including higher temperatures as biases, since both are measurable values permitted in the licensee's spent fuel pool. The staff requests the licensee amend its analysis of the Region 3 racks to include temperature as a bias.*

Response 8:

In Regions 1 and 2, the MTC is positive and the nominal design basis temperature is 150°F, the maximum permissible temperature as defined in the ANO-1 Safety Analysis Report (SAR). Because neither MCNP4a nor KENO5a is capable of calculating reactivities at 150°F, a correction factor (not a bias) was determined by CASMO4 to correct the MCNP4a calculation to 150°F. Since temperatures above the SAR limit would constitute an accident condition, credit for the soluble boron present in the pool water is assumed, as permitted by the 1978 Grimes letter and the 1998 Kopp memorandum. CASMO4 calculations have been made at temperatures above 150°F with the nominal 1600 ppm soluble boron present and the results, summarized in the attached Table R-8.1, confirm that the reactivity in both Regions 1 and 2 are well below the acceptable limit at temperatures above 150°F.

In Region 3, the MTC is negative and the design basis temperature was assumed to be 20°C. Administrative procedures will limit the minimum temperature to 20°C (68°F). Since the MCNP4a calculations are valid at 20°C, there is no correction to be made to the design basis calculation. As in the case of temperatures above 150°F, temperatures below 68°F are also considered accident conditions with credit for the soluble boron in the pool water as permitted by the NRC guidelines. This assures that the reactivity is always maintained within acceptable limits. Calculated reactivities in Region 3 for the accident condition of a temperature below 68°F (20°C) are also shown in Table R-8.1.

Table R-8.2 is attached showing a revision to the Region 3 criticality calculation.

Table R-8.1

CASMO4 Calculations of the Effect of Temperature
on the ANO Unit 1 Storage Rack Reactivities

ANO Unit 1 Region 1 with 1600 ppm Boron except Reference Case at 150°F with no Boron

Burnup (GWD/MTU)	4 C	65.5 C (150 F) (No Boron)	65.5 C (150 F)	80 C	100 C	120 C
0.000	0.91950	1.21282	0.93010	0.93295	0.93728	0.94187
5.000	0.88906	1.16674	0.89914	0.90185	0.90602	0.91044
10.000	0.86273	1.13172	0.87265	0.87532	0.87943	0.88377
20.000	0.81207	1.06856	0.82187	0.82448	0.82853	0.83280
30.000	0.76352	1.00962	0.77336	0.77597	0.78004	0.78431
40.000	0.71672	0.95288	0.72672	0.72936	0.73349	0.73782
50.000	0.67218	0.89861	0.68241	0.68510	0.68934	0.69375

ANO Unit 1 Region 2 with 1600 ppm Boron except Reference Case at 150°F with no Boron

Burnup (GWD/MTU)	4 C	65.5 C (150 F) (No Boron)	65.5 C (150 F)	80 C	100 C	120 C
0.000	0.92434	1.24114	0.93570	0.93874	0.94344	0.94839
5.000	0.89368	1.19387	0.90446	0.90737	0.91187	0.91664
10.000	0.86726	1.15826	0.87790	0.88074	0.88518	0.88986
20.000	0.81663	1.09425	0.82710	0.82988	0.83425	0.83884
30.000	0.76815	1.03450	0.77863	0.78141	0.78579	0.79038
40.000	0.72145	0.97699	0.73208	0.73488	0.73931	0.74395
50.000	0.67708	0.92199	0.68791	0.69076	0.69528	0.69999

ANO Unit 1 Region 3* with 1600 ppm Boron except Reference Case at 20 C with no Boron

Burnup (GWD/MTU)	4 C	20 C (No Boron)	20 C	80 C	100 C	120 C
0	0.82063	0.97473	0.82040	0.81751	0.81600	0.81426

* Region 3 with Fresh Fuel only.

Table R-8.2

Summary of the Criticality Safety Analyses for Storage of Fresh Fuel Assemblies in ANO Unit 1-Region 3 Racks

Reference k_{eff}	0.9740
MCNP4a Bias	0.0009
Uncertainty in MCNP4a Bias	±0.0011
MCNP4a Statistics (95/95) Uncertainty	±0.0007
Manufacturing Tolerance Uncertainty	±0.0152
Enrichment Tolerance Uncertainty	±0.0017
Depletion Uncertainty	Not Applicable
Fuel Eccentric Positioning Uncertainty	Negative
Statistical Combination of Uncertainties	±0.0154
Maximum k_{eff}	0.9903
Regulatory Limiting k_{eff}	1.0000

Question 9:

The licensee stated that the maximum k_{eff} values were calculated assuming an infinite radial array of storage cells with a finite axial length, water reflected. The staff requests the licensee specify the amount of water reflector assumed in the axial direction and justify the value chosen.

Response 9:

The analysis conservatively assumed a 30 cm water reflector top and bottom, which is effectively an infinite water reflector. The use of a water reflector thicker than 30 cm has no appreciable increase in its effectiveness as a reflector. Combinations of Zircaloy, steel, and concrete with water are less effective reflectors than pure water.

Question 10:

The licensee provided tables showing the minimum burnup required for storage of spent fuel assemblies in each of the racks as a function of cooling time and average fuel enrichment.

The staff requests the licensee specify whether the table values and the figures generated from them assumed the uncertainty in the fuel enrichment. For example, in Table 4.2.2 the burnups necessary for an enrichment of 4.95 weight percent are depicted as a function of cooling time. If the uncertainty (± 0.05 weight percent) was not considered, then the burnup or cooling times presented may be under-predicted. Longer cooling and higher burnup would be necessary to lower the reactivity to the levels calculated in the analyses. If uncertainties were not considered by the licensee in calculating the values of burnup presented, the staff requests the licensee either provide detailed technical justification for their omission or revise the tables and figures to reflect their inclusion.

Response 10:

All analyses included the uncertainty in enrichment (±0.05%); for example, see Table 4.2.1, eighth data line, with a reactivity penalty of ±0.0027 Δk, statistically combined with other data uncertainties.

Question 11:

The licensee stated that linear interpolation between the points in both Table 4.2.2 and Table 4.2.6 is acceptable since the data is "nearly" linear. Proposed Technical Specification (TS) Figures 3.7.15-1 and 3.7.15-2 provide a graphical representation of the data. Additionally, Tables 4.2.3 and 4.2.7 provide the bounding polynomial fit equations for the proposed TS figures. These equations for Region 1 and Region 2 racks are fourth and third order polynomials, respectively, implying a non-linear relationship which is seen in the TS figures. The staff requests the licensee provide information describing the following: 1) the basis for assuming a linear relationship; 2) the maximum error that can be introduced by assuming a linear relationship; 3) how this error was accounted for in the criticality analyses; 4) how the error will be limited when using the TS figures; and 5) the effects of assuming a conservatively bounding second or first order polynomial.

Response 11:

Please see the response to question 5.

Question 12:

The licensee's criticality analysis has identified the misloading of a fresh fuel assembly into a Region 2 cell intended to remain empty as an event which requires 800 ppm of soluble boron to assure the max k_{eff} does not exceed 0.95. The staff requests the licensee identify controls which either will be put into place or are already in place to prevent this event from occurring.

Response 12:

Prior to moving a fuel assembly, the fuel assembly is required to be classified based on fuel enrichment, decay time, and burnup to determine where it can be placed in the SFP. ANO administratively controls the proper placement of fuel assemblies based upon their fuel assembly classification. A qualified engineer using the procedure for controlling special nuclear material checks the fuel assembly classification for each assembly. An independent review by a qualified engineer assures the fuel assembly classification.

The above defined classification process is used before placement of fuel assemblies into either Region 1 or Region 2. The process for placement of fuel assemblies within the pool is controlled by site procedures as described in the next paragraph.

The placement of fuel assemblies within proposed Region 3 will be unrestricted. If it is desired to move a fuel assembly to Region 1 then the classification of that fuel assembly for that region is checked. If the fuel assembly is unrestricted for Region 1 then that assembly can be stored unrestricted within Region 1. If the fuel assembly is restricted for Region 1 then the assembly will be stored in a checkerboard arrangement. The same logic is used to determine placement of an assembly in Region 2. The qualified engineer then develops a special nuclear material transfer report. The engineer checks the most recent inventory map, any restricted storage areas, and checks any outstanding transfer reports that have been performed after the last update to the map. The qualified engineer verifies that the cell where the fuel assembly will be stored is empty and the cell is located in the correct region. If a checkerboard loading pattern is required, then the qualified engineer verifies that the surrounding cells where the restricted fuel assembly will be stored meet the checkerboard loading pattern requirements. The area that is designated to require a checkerboard pattern is classified as a checkerboard area and the empty spaces are procedurally controlled. If required, the qualified engineer ensures assemblies are moved out of cells that are required to be empty of special nuclear material prior to placing a restricted fuel assembly in the checkerboard arrangement. The qualified engineer will complete the transfer report.

The entire process for controlling the movement of fuel assemblies is independently reviewed by another qualified engineer. The reactor engineering superintendent approves the report and ensures that the report has been independently reviewed. The transfer report is then sent to operations for review.

Movement of fuel assemblies is performed by qualified fuel handling personnel. Prior to grappling and un-grappling a fuel assembly the qualified fuel handler verifies the fuel handling grapple or fuel assembly is over that correct location designated by the transfer report. These locations are independently verified prior to grappling and un-grappling the assembly. The fuel movement process is supervised by an operations designee or by a qualified engineer. Upon completion of the special nuclear transfer report a qualified engineer reviews the report and verifies the fuel moves were completed satisfactorily. Two qualified engineers then independently perform a survey of the storage locations to verify that the special nuclear material has been moved to the proper location. After the special nuclear material location verification is complete, the inventory maps are updated along with the special nuclear material location record.

Question 13:

The licensee described a limitation of the MCNP calculations which prevented the modeling of some fission product nuclides in the criticality analyses. The licensee described a process to calculate an equivalent amount of boron which provides nearly the same reactivity in MCNP as the CASMO4 result. The licensee stated this would compensate for *the inability of MCNP to model these nuclides which account for approximately one percent of the reactivity. The staff requests the licensee provide detailed technical information demonstrating that this alternate methodology is conservative or provides bounding results. Additionally, the staff requests the licensee provide a table of the nuclides not modeled in the MCNP correlation and a quantitative summary of the equivalent boron-10 used to account for their reactivity.*

Response 13:

Neither MCNP4a nor NITAWL-KENO5a has all of the fission product cross-sections in their applicable libraries. CASMO4 tracks the concentrations of the most important 49 actinide and fission product nuclides. Fission product nuclides that are not tracked in CASMO4 are collected together and described by two pseudo-fission nuclides, called LFP1 and LFP2. In addition to the two pseudo-fission products, MCNP4a and KENO5a do not have in their libraries the following six nuclides:

U-239	Np-239	Ba-140
La-140	Pm-148m	Eu-148

Of these 6 nuclides, only Pm-148m is significant and for conservatism the remaining 5 nuclides (together with Xe-135) are conservatively set to zero concentration. There are 3 nuclides, which have no cross-section libraries in either MCNP or KENO5a, Pm-148m and the 2 pseudo-fission products. For the past several years, in many licensing applications reviewed and accepted by the NRC, it has been standard practice to calculate an equivalent boron-10 concentration to compensate for the absence of cross-sections for those nuclides in the MCNP4a and KENO5a libraries. The only known alternative methodology is the reactivity-equivalent methodology, which was not used for the study supporting the licensee amendment. (In a recent study, NUREG CR/6683, ORNL has submitted results of some calculations that challenge the validity of KENO calculations, although MCNP4a calculations remain accurate.)

To illustrate the application of the compensating B-10 concentration, the Unit 1 Region 1 analysis at 43 MWD/KgU burnup and 0 cooling time is used as an example. For this condition, the concentrations of the 3 fission products (including the lumped fission products) and the equivalent B-10 are as follows:

Pm-148m	6.73318E-8 atoms / b-cm	(Δk = +0.0020)
LFP1	1.15054E-3 atoms / b-cm	(Δk = +0.0040)
LFP2	2.49288E-4 atoms / b-cm	(Δk = +0.0079)
Equivalent B^{10}	4.0801E-6 atoms x 10^{-24}/ b-cm	(Δk = -0.0139)

Check calculations gave the following results:

MCNP4a cell	k_{inf}= 0.9621 $\pm$ 0.0013 (95%/95%)
Reference CASMO4	k_{inf}= 0.9637

In the rack calculations, the analyses shown in the example above are repeated for each of the 10 axial zones with burnups according to the axial burnup distribution to obtain the final calculated k_{eff} value.

Question 14:

The licensee's accident analyses did not include a discussion of the effects of pool temperatures greater than 150 °F. The positive MTC in Regions 1 and 2 will cause a reactivity addition when pool temperatures increase. The staff requests the licensee analyze this event and provide a detailed analysis of the results.

Response 14:

Please see the response to question 8 for the discussion of the effects of temperature and the relationship to accident conditions.

Question 15:

For the most limiting dropped fuel assembly analysis, the licensee assumed that all of the Metamic® poison panels would be lost in Region 3. The licensee identified the need for a soluble boron concentration of 1600 ppm to maintain k_{eff} at or below 0.95. The staff requests the licensee provide a list of assumptions and their justifications used in the analysis. Examples of the information the licensee should provide include the following: a) the type of assembly dropped (fresh, spent, burnup, enrichment, cooling time, etc.); b) the loading of the rack during the accident, and c) the temperature of the spent fuel pool.

Response 15:

The assumption that all the Metamic in Region 3 would be stripped off by the drop of a single fuel assembly is an extremely conservative and unrealistic scenario, chosen solely to illustrate the very large margin of safety. The spent fuel pool normally contains at least 1600 ppm soluble boron and both the April 1978 Grimes letter and the 1998 Kopp memorandum allow full credit for the soluble boron present (double contingency principle). Furthermore, the Metamic panels are protected by the rack structure and cannot be contacted by the dropped assembly in more than the immediately affected 4 cells. The analysis in the licensing report assumed that the dropped assembly and all the assemblies in the rack are fresh, unburned fuel of the maximum permissible enrichment (4.95 %). The reference temperature of 20 °C was used for this analysis.

Question 16:

The licensee specified that a loss of coolant at a rate of 2 gallons per minute (gpm) with a makeup of 2 gpm of unborated water would dilute the spent fuel pool to a concentration of 400 ppm in 123 days under normal operating conditions. The staff requests the licensee identify the flow rate which would dilute the pool to less than 400 ppm within 7 days. Additionally, the staff requests the licensee identify all means which could provide this rate of inventory loss and describe the controls in place to limit the potential for their occurrence.

Response 16:

35.3 gpm would be required to dilute the soluble boron concentration from 1600 (T.S. 3.7.14) to 400 ppm in seven days.

The above event requires 356,000 gallons to dilute the soluble boron concentration from 1600 to 400 ppm. A volume loss of this magnitude would require at least two or more independent failures and therefore is not credible. In addition if multiple failure occurred, leakages of this magnitude would be readily evident through control room level and radiation annunciators and/or plant walk downs. SFP addition to makeup for any losses is only by manual operator action and procedurally controlled to prevent dilution.

Question 17:

In reviewing Tables 4.2.1, 4.2.4, 4.2.5, 4.2.8, and 4.2.9 of the licensee's submittal, the staff identified differences in the reactivity effect of the manufacturing tolerance uncertainty between the analyses. The differences appear not only between regions but within one region when the spent and fresh fuel analyses are compared. The staff requests the licensee describe and justify the reasons for the differences.

Response 17:

It is expected that the reactivity effects of manufacturing and other tolerances will differ between regions due to different designs, burnups, and/or operating conditions. These reactivity effects (uncertainties) have been independently evaluated for the numerous designs and incorporated into Tables 4.2.1, 4.2.5, 4.2.8 and 4.2.9.

The Region 1, 2 and 3 racks have different designs. Furthermore, the required spent fuel burnups differ between Regions 1 and 2 which results in differences in the neutron spectra that must be considered in evaluating reactivity effects of tolerance. Region 3 is designed for unrestricted storage of fresh fuel assemblies and employs Metamic as a neutron absorber. These differences also result in different reactivity effects of tolerances that must be independently evaluated. The net effects of these design and fuel characteristics (burnup) are the differences in reactivity effects of various tolerances.

Question 18:

In each of the analyses presented in Tables 4.2.1, 4.2.4, 4.2.5, 4.2.8, and 4.2.9, the licensee includes a term identified as "MCNP4a Statistics (95/95) Uncertainty." The staff requests the licensee describe, in greater detail, how this value is obtained. Additionally, the staff requests the licensee include relevant equations and numerical data which demonstrate how the value was calculated.

Response 18:

The evaluation of the MCNP4a statistics (95/95) was based on the conventional technique described in NBS (now NIST) Handbook 91, *Experimental Statistics*, by M.G. Natrella. This technique requires that the standard deviation of the MCNP4a calculated k_{eff} be multiplied by a factor (commonly known as k-factor) for 95% probability, 95% confidence level as provided in NBS Handbook 91. The same technique would also apply to KENO5a calculations. For example, the standard deviation of k_{eff} in the MCNP4a calculations is usually ±0.0004 Δk. For the large number of generations (degrees of freedom) in the MCNP4a calculations, the multiplier for a one-sided tolerance at the 95% probability/95% confidence level is 1.67 from the NBS Handbook 91. Increasing this factor to 1.7 for conservatism, the MCNP4a statistics (95/95) becomes 1.7*0.0004=0.0007 Δk which is the value listed in the Tables and incorporated into the analyses.

Question 19:

The licensee's application does not discuss the interfaces which may occur between regions, racks, or within a rack. The staff requests the licensee identify any interfaces which may occur with the proposed configurations, such as a fresh fuel assembly in a Region 1 rack adjacent to a fresh fuel assembly in a Region 2 rack, or a spent fuel assembly adjacent to a fresh fuel assembly either in the same rack or adjacent racks. The staff requests the licensee provide a response to either of the following concerns.

a. *If these types of interfaces are possible but bounded by other normal conditions described in the amendment, the staff requests the licensee provide a detailed justification of the basis for claiming the interface conditions are bounded.*

b. *However, if these types of interfaces are possible but not bounded by other conditions described in the amendment, the staff requests the licensee provide a detailed quantitative analysis of the reactivity effects associated with the interface conditions.*

Response 19:

The large (neutronically) water gaps between modules and between regions sufficiently offsets reactivity interactions across interfaces. Therefore, the normal conditions of storage yield reactivities that bound or equal all such surfaces. To provide confirmation of this, specific MCNP4a calculations were made for all recognized interfaces and the results are summarized in Table R-19.1 attached. The calculational models were based on the region configurations and rack-to-rack spacing. These calculations confirm that all possible interfaces have reactivity effects that are well within the regulatory limit.

Region 1 has no interface with Region 2. Calculations shown in Table R-19.1 evaluate a Region 1 to Region 3 interface or a Region 2 to Region 3 interface. The Region 3-Region 1 interface (with spent fuel in Region 1 and fresh fuel in Region 3) yields a k_{eff} slightly higher than the normal configuration. The same conclusion results for interfaces with fresh fuel assemblies on one side (Region 3) and spent fuel assemblies in the adjacent rack (Region 1 or 2).

There is the potential for fresh fuel assemblies being stored (checkerboard pattern) in the same module adjacent to an array of spent fuel assemblies with no intervening water gap. Results of such calculations for both Region 1 and Region 2, confirm that the reactivity is less than and bounded by the limiting reactivity of the spent fuel infinite array in the reference calculations.

Table R-19.1 illustrates that the normal water gaps between modules and between Regions would not result in reactivity exceeding the regulatory limit.

Table R-19.1 Summary of Interface Calculations

Configuration	Calc. k-eff	Bias/uncertainties	Max k-eff [1]	Ref k-eff [2]
A) Region 1 rack to rack interface w/ adjacent fresh fuel assemblies in both modules (Abnormal Configuration).	0.9256	0.0179 Table 4.2.4	0.9435	0.9443 Table 4.2.4
B) Region 1 with fresh fuel in checkerboard pattern and spent fuel in the same rack module – no intervening water-gap.	0.9607	0.0240 Table 4.2.1	0.9847	0.9950 Table 4.2.1
C) Region 2 rack to rack Interface with adjacent fresh fuel assemblies in both modules (Abnormal Configuration).	0.9482	0.0262 Table 4.2.8	0.9744	0.9621 Table 4.2.8
D) Region 2 with fresh and spent fuel in the same module – no intervening water-gap.	0.9602	0.0309 Table 4.2.5	0.9911	0.9928 Table 4.2.5
E) Region 1 and Region 3 rack to rack interface with fresh fuel checkerboard pattern in Region 1. Adjacent fresh fuel assemblies across the water-gap.	0.9719	0.0163 Table 4.2.9	0.9882	0.9903 Table 4.2.9
F) Region 1 – Region 3 Interface with spent fuel in Region 1 and fresh fuel in Region 3.	0.9721	0.0240 Table 4.2.1	0.9961	0.9950 Table 4.2.1
G) Region 2 and Region 3 rack to rack Interface with fresh fuel in both regions. Checkerboard pattern in Region 2.	0. 9582	0.0163 Table 4.2.9	0.9745	0.9903 Table 4.2.9
H) Region 2 – Region 3 Interface with spent fuel in Region 2 and fresh fuel in Region 3.	0.9592	0.0309 Table 4.2.5	0.9901	0.9928 Table 4.2.5
I) Region 3 with fresh fuel in both racks.	0.9760	0.0163 Table 4.2.9	0.9923	0.9903 Table 4.2.9

(1) The maximum k-eff includes biases and uncertainties associated with the higher reactive (dominating) region.
(2) Maximum reactivity of the higher reactivity region.

Question 20:

The licensee provided a description of the current licensing basis for its spent fuel pool which is based on an exemption to 10 CFR 70.24, "Criticality accident requirements." The licensee has chosen to change its regulatory basis to comply with 10 CFR 50.68. Since the amendment proposes a change in the regulatory basis to which ANO-1 has not previously demonstrated compliance, the staff requests the licensee provide a summary of how it will comply with all parts of 10 CFR 50.68 paragraph (b) if its amendment receives approval.

Response 20:

ANO-1 will comply with the requirements of 10 CFR 50.68. This regulation requires the following:

50.68(b)(1) - Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

It has been determined that movement of only one fuel assembly at a time assures that subcriticality is maintained under the most adverse moderation conditions feasible by unborated water. ANO fuel handling procedures for the SFP and reactor refueling bridges exclusively prohibit the movement of more than one fuel assembly over the SFP or the refueling canal. Movement of a fuel assembly using the upender frame is allowed while the fuel handling bridges are moving fuel assemblies because it has been determined that for the worst case geometry k-effective is less than 0.95 at a 95% probability with a 95% confidence level. The fuel receipt procedure only allows one new fuel assembly to be moved at a time. When new fuel assemblies are moved from the new fuel vault to the new fuel elevator, which is located in the tilt pit, the spent fuel bridge grapple is procedurally required to be located outside the tilt pit. Only one fuel assembly at a time is procedurally allowed to be moved into a dry fuel storage cask.

Storage of fuel assemblies is procedurally controlled to assure k-effective remains below 1.0, at a 95% probability, 95% confidence level, when flooded with unborated water. The storage patterns assure subcriticality under the most adverse moderation conditions by unborated water. The storage patterns will also insure reactivity will not exceed 0.95 at a 95% probability with a 95% confidence level when credit is taken for 400 ppm boron.

50.68(b)(2) - The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

Criticality calculations have been performed on the new fuel vault fully loaded with B&W 15x15 fresh fuel assemblies and filled with the most reactive unborated water.

The results of these calculations showed that reactivity did not exceed 0.95, at a 95% probability, 95% confidence level.

50.68(b)(3) - If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

Criticality calculations were performed on the new fuel vault fully loaded with B&W 15X15 fresh fuel assemblies and filled with the most reactive low density hydrogenous fluid. The results of these calculations showed that reactivity did not exceed 0.98, at a 95% probability, 95% confidence level. Hydrogenous fluid are not used in the new fuel vault area and they would only be used in the most extreme cases where the use of fire water was not able to contain a fire in the new fuel pit area.

50.68(b)(4) - If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

Soluble boron credit will be taken in the spent fuel pool storage racks. The criticality calculations included in the proposed change show that k-effective remains below 1.0, at a 95% probability, 95% confidence level, when flooded with unborated water. Reactivity will not exceed 0.95 at a 95% probability with a 95% confidence level when credit is taken for 400 ppm boron.

50.68(b)(5) - The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

Any quantity of SNM that is received on site will be tracked to ensure that the total quantity remains less than that needed to form a critical mass.

50.68(b)(6) - Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

A radiation monitor is located in the Unit 1 SFP area and alarms within the control room. During fuel movement activities additional radiation monitors are located directly on the fuel handling bridges to provide an additional audible indication of excessive radiation levels. The fuel handling procedures require the additional radiations monitors to be in place when fuel movement is in progress.

50.68(b)(7) - The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

The proposed change to the ANO-1 Technical Specifications (TS) 4.3.1.1.a and 4.3.1.2.a will limit the maximum U-235 fuel enrichment to 5.0 weight percent including enrichment uncertainties.

50.68(b)(8) - The FSAR is amended no later than the next update which 50.71(e) of this part requires, indicating that the licensee has chosen to comply with 50.68(b).

The FSAR will be amended no later than the next required update after the proposed TS change is approved and implemented. This FSAR update will indicate that ANO has chosen to comply with 50.68(b).

Question 21:

The licensee proposed two enrichment-dependent storage patterns for the fresh fuel storage racks. These storage patterns define which cell locations within the racks must remain empty for a fresh fuel assemblies at enrichments of 4.2 weight percent or 5 weight percent. The staff requests the licensee describe whether it intends to store fresh fuel assemblies of different enrichments within the same rack and if so, how it will control the loading pattern under those conditions.

Response 21:

The new fuel vault is intended to store either 5% (Figure 4.10.2) or 4.2 wt. % (Figure 4.10.3) in different arrangements. If any of the fuel assemblies stored in the new fuel vault have enrichments greater than 4.2 wt. %, then the assemblies will be stored in accordance with Figure 4.10.2. Only if all assemblies stored contain fuel of 4.2 wt. % enrichment or less, would the configuration depicted in Figure 4.10.3 be applicable.

Question 22:

The licensee's amendment request proposes not to credit Boraflex currently installed in the Region 1 racks for criticality control. To support this, the licensee analyzed the Region 1 racks assuming the Boraflex was not present. The staff requests the licensee identify how the current Boraflex panel was modeled in the criticality analysis. Additionally, the licensee identified that the Region 1 racks will have a positive moderator temperature coefficient. However, the presence of Boraflex could result in a negative moderator temperature coefficient. The staff requests the licensee describe how it verified that a positive moderator temperature coefficient is present in the Region 1 racks.

Response 22:

Analyses of the Region 1 racks were made with the assumption that the Boraflex absorber is completely lost. Without any Boraflex, the analyses confirmed that the temperature coefficient of reactivity is positive. Any residual Boraflex that might exist, now or in the future, would tend to reduce the reported maximum reactivity and to reduce the magnitude of the positive temperature coefficient of reactivity. If a large fraction of the initial Boraflex remains, the temperature coefficient of reactivity would be negative and the reported maximum reactivity would be even more conservative. Thus, any Boraflex remaining in the Region 1 rack would make the reported analyses conservative.

Question 23:

In sections 4.5.1, 4.6.1, 4.7.1, and 4.8.3 of Holtec Report, HI-2022867, the licensee stated that the regulatory limit of k_{eff} in the spent fuel pool was "[less than or equal to] 1.0". However, 10 CFR 50.68 states that the regulatory limit, when no credit for soluble boron is taken, is "[less than] 1.0". The staff requests the licensee review the sections listed and reconcile the differences.

Response 23:

The reviewer is correct. The phrase "or equal to" is incorrect and has been removed.

Attachment 2

To

1CAN110302

**Response to Materials and Chemical Engineering Branch
Request for Additional Information**

**Response to Materials and Chemical Engineering Branch
Request for Additional Information
Related to Changes to the Spent Fuel Pool Loading Restrictions**

Question 1:

The submittal references a topical report submitted to the staff on August 8, 2002. This topical report describes the physical and chemical properties of Metamic® as well as the test results supporting the use of Metamic® in fuel pool applications. The staff notes the types of coupons tested and discussed in that report; i.e., 15 weight percent (wt %) or 31 wt % B_4C, mill-finished or anodized.

Based on the information provided in the submittal, the staff requests the licensee to address the following:

a. *Identify if the 25 wt% B_4C Metamic® poison inserts to be used are mill-finished or anodized.*

b. *The topical report concludes that corrosion on both mill-finished and anodized Metamic® coupons is due to inadequate cleaning of the surface. Discuss the cleaning technique to be used on the inserts prior to installation, its acceptability, and its expected effectiveness in controlling impurities.*

c. *If anodized inserts are to be used, discuss the anodizing process used for these inserts and its effectiveness in reforming the protective oxide layer in the event the surface should be scratched. Discuss the process used to ensure that the protective oxide layer will be formed adequately during installation.*

Response 1:

a. Based on the environmental conditions attendant to Metamic in wet storage, the product is provided in mill-finished condition. Tests show that the mill-finished Metamic develops a stable, hard, and continuous Aluminum Oxide (Al_2O_3) coating.

b. The EPRI topical report is correct in positing that inadequate cleaning can be a source of surface contamination that can cause pitting corrosion of Metamic in long-term use under certain environmental conditions. The principal sources of nucleation sites for pitting has been determined to result from the high temperature extrusion process wherein a large force is used to push the heated sintered billet through the die. Abrasion of the die by the hard boron carbide particles is the principle source of ferritic particulates that become lodged on the surface of the extruded work piece. This source of iron has been eliminated in the manufacturing of Metamic by introducing a post-extrusion cleaning step wherein the extruded billet is subjected to air-blasting with glass beads or hard alumina powder, which is effectively a mechanically forced erosion process. The subsequent operation, namely, rolling of the extruded bloom stock and shearing, results in minuscule levels of iron remaining on the surface of the finished Metamic panels. Preliminary tests on Metamic manufactured with post-extrusion glass beading showed absence of any significant pitting corrosion after an extended period of soaking (forty-five days) in heated water (at approximately 170°F). The test report will be finalized after the neutron attenuation test data can be fully analyzed. After the rolling and sheering processes, on an as needed basis the Metamic panels will be cleaned by an appropriate process (glass-beading, aluminum powder blasting, or cleaning with a dilute nitric acid rinse). Finally, the Metamic panels are wiped clean of all non-adherent dust, grease, and the like using denatured alcohol.

To provide an additional confidence that the Metamic panels will render their intended function, randomly drawn test coupons from each batch will be subjected to a thirty-day high temperature water soak test and will be examined for their physical integrity, surface condition, and dimensional stability. In addition, surveillance coupons, mounted on a "tree" will be placed in the fuel storage pool for long-term surveillance to monitor (and confirm) the performance of Metamic under the actual radiation, chemical and temperature conditions in the spent fuel pool.

c. Anodizing is not used.

Question 2:

The Metamic® coupon sampling program is briefly discussed on page 5 of Attachment 1 to the submittal. The licensee commits to establishing this program to monitor the physical and chemical properties of Metamic® over time. Metamic® coupons will be suspended on a mounting tree and inserted into an empty fuel cell in a rack that is surrounded by spent fuel assemblies. A total of 10 coupons will be created from the same lots that will be used to manufacture the inserts.

Metamic® is a new material to be used in the spent fuel pool environment. While the staff agrees that a coupon sampling program is prudent and critical in verifying the assumptions used in the spent fuel pool criticality analysis, details of this program were not provided in the submittal. Therefore, the staff requests that the licensee discuss the following:

- *size and types of coupons to be used; i.e., similar in fabrication and layout to the proposed insert including welds and proximity to stainless steel;*
- *technique for measuring the initial B_4C content of the coupons;*
- *simulation of scratches on the coupons;*
- *frequency of coupon sampling and its justification; and*
- *tests to be performed on coupons; e.g., weight measurement, measurement of dimensions (length, width and thickness), and B_4C content.*

Response 2:

A coupon surveillance program will be implemented to maintain surveillance of the Metamic absorber material under the radiation, chemical, and thermal environment of the SFP. This program will include the following:

- The surveillance coupons will be approximately 7" x 5" and 0.100" thick, identical in composition and manufacturing process as the Metamic in the inserts. The coupons will be mounted in stainless steel jackets simulating the actual insert design. The coupon tree will have ten or more coupons. To better simulate the conditions of the

Metamic panels, the coupon tree has been designed to be installed within a flux trap. The coupon tree will be located in the Region 2 racks. The coupons will be staggered and placed adjacent to the active fuel region where, based on the burnup profile, the localized bumup is greater than the assembly average burnup. This design will maximize the dose received by the coupons as compared to the Metamic inserts and accurately simulate the flow characteristics, pool chemistry, and differential metal interfaces that the Metamic inserts will experience. No welding will be used on the Metamic as per the insert design.

- The initial B_4C content of the surveillance coupons will be determined by the same chemical analysis technique used to establish the B_4C content of the Metamic in the insert. The B_4C used in the production of the Metamic is tested to determine the concentration of boron within each batch. The B_4C and aluminum matrix are accurately weighed and combined in a ratio that will assure the panels have a B_4C content higher than 24.5 weight percent. Samples of the B_4C and aluminum mixture are then chemically tested to determine the B_4C concentration. Additional samples are then taken of the final product, which are also chemically tested to confirm the concentration of B_4C in Metamic. Representative samples of the batches used will undergo neutron attenuation testing to validate the B10 loading in the Metamic panels and coupons.

- Scratches will be simulated by the mechanical etching or scribing the surface of the coupon.

- The surveillance coupons will be removed and examined on a regular schedule to be established by a written procedure. Coupons will be examined on a two year basis for the first three intervals and thereafter at increasing intervals over the service life of the inserts. During the first six years, freshly discharged fuel assemblies will be placed on two sides of the coupon tree to ensure that the dose to the coupons is maximized. The initial two year period is based on providing sufficient exposure time to the environment such that any degradation would be observable.

- Measurements to be performed at each inspection will be as follows:

  A) Physical observations of the surface appearance to detect pitting, swelling or other degradation,
  B) Length, width, and thickness measurements to monitor for bulging and swelling
  C) Weight and density to monitor for material loss, and
  D) Neutron attenuation to confirm the B-10 concentration or destructive chemical testing to determine the boron content.

Attachment 3

To

1CAN110302

**Response to Plant Systems Branch
Request for Additional Information**

**Response to Plant Systems Branch
Request for Additional Information
Related to Changes to the Spent Fuel Pool Loading Restrictions**

Question 1:

In the staff's safety evaluation for WCAP-15516-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," the staff stated that all licensees proposing to credit soluble boron should identify potential events which could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k-eff limit and should quantify the time span of these dilution events to show that sufficient time is available to enable adequate detection and suppression of any dilution event. The staff also stated that the effects of incomplete boron mixing, such as boron stratification, should be considered, and that the boron dilution analysis should also be used to justify the surveillance interval used for verification of technical specification minimum pool boron concentration. In order to complete our review, the NRC staff requests that Entergy Operations provide the following information regarding Section 4.9, "Soluble Boron Dilution Evaluation," of Attachment 4 to their letter dated April 2, 2003:

a. *The dilution evaluation considered a long term loss of inventory at a rate of 2 gpm, but justification for this value as bounding for a short-term uncorrected condition was not adequate. Recognizing that evaporative losses may be very small at times of low pool temperature and some low-level of leakage may be accepted without corrective action, provide justification that a loss of inventory of this magnitude would be promptly corrected. This response should include expected alarms, the expected change in frequency of these alarms, and specific steps from alarm response or off-normal procedures that would lead to corrective actions to stop the inventory loss.*

b. *Briefly discuss whether damage to the spent fuel pool floor resulting from light load drops would cause a loss of coolant inventory in excess of the capacity of the normal makeup systems. If considered, describe how this event and the potential dilution resulting from use of unborated makeup water sources would be detected and mitigated. This response should include expected alarms and specific steps from alarm or off-normal procedures that would lead to corrective actions.*

Response 1:

a. Technical Specification (TS) Surveillance Requirement (SR) 3.7.13.1 requires monitoring of SFP level. To satisfy this TS SR, the ANO-1 operations staff monitors SFP level daily using a Control Room indication and weekly locally at the SFP. The operations staff is trained to monitor for unexpected deviations in their recorded readings and take appropriate actions to determine and correct the cause of the deviation as needed. In the unlikely event the decreasing trend is not observed, a SFP low level annunciator is provided in the control room which alarms at 6" below normal pool level. Upon receipt of the alarm, an operator is dispatched to the SFP to verify actual level and if level is dropping, the SFP cooling pumps are secured, if necessary, and SFP level is adjusted to within limits. If the SFP level alarm malfunctioned at this magnitude of inventory loss, it would take a long period of time for the SFP level to significantly drop (the SFP contains over 600 gallons/inch). Eventually, there would be an increase in radiation levels that would be seen by the

SFP area radiation monitor which is monitored every 12 hours and alarms in the control room. Additionally, the SFP boron concentration is sampled at least weekly as required by TS SR 3.7.14.1. If there were an unexpected downward trend in boron concentration, operations would be notified to investigate the cause and take the appropriate correction actions.

b. A loss of coolant inventory in excess of the capacity of the normal makeup systems is not expected to occur in the unlikely event of a light load drop. Analyses indicate that the drop of the SFP gate will result in no significant damage to the 5'6" slab.

Leakage, although unexpected, would be indicated by receipt of the control room SFP low level annunciator. The control room staff would, as directed by procedure, visually verify actual level and commence makeup to the spent fuel pool as necessary. If unborated water sources are used for makeup, which is a manual operation, then chemistry samples are taken to ensure the SFP boron concentration is within the requirements of Technical Specifications.

Question 2:

Section 2.0, "Proposed Change," of the Entergy licensing amendment request, dated April 2, 2003, states the following:

> *"ANO-1 TS 3.7.14 and Surveillance Requirement 3.7.14.1 define the minimum required SFP boron concentration. The proposed change will result in changing the "greater than or equal to" sign to a "greater than" sign. The proposed change, though minor, results in a more accurate representation of the analysis assumption of greater than 1600 parts per million (ppm) for the boron concentration."*

Briefly discuss the rationale for using the "greater than" 1600 ppm for the SFP boron concentration; is there an upper bound? Does use of the "greater than" boron concentration enhance the capability to mitigate boron dilution events?

Response 2:

A boron concentration of 1600 ppm preserves the assumptions used in the analyses of the potential accident scenarios as does "greater than" 1600 ppm. The "equal to" sign was removed to be consistent with the ANO-2 TSs. The use of the "greater than" boron concentration does not enhance the capability of mitigation of boron dilution events. The upper limit on boron concentration is 3500 ppm per the ANO-1 Safety Analysis Report, Section 9.6.2.4.3.4.

Question 3:

Describe features of the spent fuel pool cooling system which ensure adequate boron mixing in the SFP and thereby, prevent or minimize the occurrence of boron stratification during a boron dilution event.

Response 3:

The two spent fuel pool circulating pumps take suction from the SFP and recirculate the fluid back to the pool after passing through the SFP coolers. At least one pump is normally in service. Cooler water is discharged into the pool through two nozzles simultaneously. One of the nozzles is located near the water surface while the other one is near the bottom of the pool. The suction nozzle for the SFP circulating pumps is located on the opposite end of the pool from the discharge nozzles and is near the water surface. This arrangement provides thermal mixing and boric acid mixing.

Significant convective currents are also created from heat being transferred out of the stored spent fuel assemblies which contributes to the mixing in the pool.

Attachment 4

To

1CAN110302

Response to Mechanical and Civil Engineering Branch
Request for Additional Information

**Response to Mechanical and Civil Engineering Branch
Request for Additional Information
Related to Changes to the Spent Fuel Pool Loading Restrictions**

Question 1:

The first paragraph of page 6-3 of the Holtec report (Attachment 4) states that 3-D Whole Pool Multi-Rack (WPMR) analyses carried out on several previous plants demonstrate that single rack simulations may under predict rack displacement during seismic response. Is the nature of the above mentioned "under-prediction in rack displacement" with respect to a set of closed form solutions or responses obtained via pertinent tests? If yes, briefly describe the applicable closed form solutions or tests conducted, otherwise, explain the basis of your statement.

Question 2:

Regarding the second paragraph of page 6-4 of the report, discuss the laboratory experiments that were conducted to validate the multi-rack fluid coupling theory and the basis for specific coupling parameters used in the ANO-1 WPMR analyses.

Question 3:

The second paragraph of page 6-4 of the report states that the WPMR analyses have corroborated the accuracy of the single rack 3-D solutions in predicting the maximum structural stresses, and served also to improve predictions of rack kinematics. Explain the basis of this statement including to what specific "accurate 3-D solution" the statement is meant to apply. Also, indicate if ANO-1 has a set of known or true rack kinematic data to serve as a reference solution for the statement.

Question 4:

Referring to the last paragraph of page 6-4 and the last paragraph on page 6-20 of the report, it is not clear as to whether both the single rack and the WPMR analyses were used by ANO-1 for the LAR. Discuss how the results of these analyses were utilized in providing an adequate seismic/structural analysis basis for the ANO-1 spent fuel rack related LAR. Also, describe how Case 7 on page 6-21 relates to the proposed license amendment.

Question 5:

With respect to item d. of page 6-8 of the report addressing fluid coupling effect, discuss any pertinent experimental work performed after 1982 which supports the adequacy and reasonableness of the methods reported in references 6.5.2 and 6.5.3. Also, the second paragraph of page 6-11 of the report states that the derivation of fluid coupling matrix has been extensively verified by an extensive set of shake table experiments (Ref. 6.5.5, proprietary). Briefly summarize the experimental results that are relevant to ANO-1 WPMR analyses.

Question 6:

Regarding Section 6.8.4.3, "Fuel to Cell Wall Impact Loads," of page 6-23, discuss briefly how the evaluation of cell wall integrity under the instantaneous impact of fuel assembly load was performed to conclude that cell walls will remain intact with no permanent damage. Additionally, referring to the last paragraph of page 6-25 of the report, discuss briefly the basis for obtaining the limiting impact load of 2436 lb/ft for rack cell wall.

Question 7:

Figure 6.11.1 of the report shows maximum instantaneous hydrodynamic pressures for both the OBE and the SSE that are indicated to be applicable to spent fuel pool walls adjacent to a rack. Are these data ever corroborated by applicable tests and are the negative wall pressures shown in the figure realistic? Briefly discuss how this figure is applied specifically to the integrity evaluation of ANO-1 spent fuel pool structures.

Question 8:

Referring to page 6-13 of the report discussing Rabinowicz's data and fixing the randomly selected coefficient of friction for each simulation, briefly summarize Rabinowicz's experiments and elaborate on the meaning of a statement that the coefficient of friction between the pedestal supports and the pool floor is indeterminate. Also indicate the minimum number of simulations for the friction coefficient performed (refer to the footnote of the page) in a given WPMR analysis to ensure that the results obtained are meaningful and discuss the basis for selecting the minimum number.

Question 9:

For the plants listed in Table 6.2.1, discuss briefly, as applicable, the actual operating experience of plants whose spent fuel racks/spent fuel pools have been subjected to real earthquakes of some significance and displayed seismic responses appreciably different from those obtained from DYNARACK analyses.

Question 10:

Section 6.10 (page 6-28) of the report states that an evaluation will be performed to demonstrate that the stresses in the poison insert, under normal and accident conditions, meet the appropriate stress limits from ANO Specification No. AL&L-C-502 and the ANO Unit 1 SAR, and that the minimum calculated safety factor for the poison insert for all loading conditions will be greater than 1.0. In the submittal, you commit to complete this evaluation within 90 days of approval of the proposed change. Briefly describe how the evaluation will be performed, and discuss the design and acceptance criteria in the above documents.

Response 1-10:

Due to the redesign of the stainless steel frame, which provides the support structure for the Metamic panels, a new structural analysis must be performed. Stevenson & Associates, who is knowledgeable in seismic structural design, will independently re-perform the spent fuel pool rack structural/seismic analysis and the poison insert design qualification. As part of the reanalysis work, questions 1 through 10 will be considered. However, the reanalysis will be performed using more classical methods with commercially available codes (i.e. ANSYS). Section 6.0 of the licensing report that accompanied the licensing submittal will be replaced entirely to reflect the results of the re-performed analysis.

Question 11:

With respect to Attachment 5, "Evaluation of Spent Fuel Pool Structural Integrity for Increased Loads from Spent Fuel Racks," discuss past operating, inspection and maintenance experience of the ANO-1 spent fuel racks and spent fuel pool structures. As applicable, discuss any pool wall/bottom slab cracking, settlement and/or pool water leakages observed to date and repair measures taken to remedy them.

Response 11:

SFP structural integrity is monitored under the Maintenance Rule structural monitoring program established in ANO Engineering Standard CES-19. This standard requires periodic condition monitoring of key structures at ANO including the spent fuel storage structure. The scope of the inspection includes, but is not limited to, visual examination for any evidence of cracks wider than 1/16", spalling, scaling, stratification, water infiltration, rust bleeding, exposed or corroded rebar, discoloration, concrete erosion or chemical attack. The initial baseline inspection of the ANO-1 SFP structure identified no deficiencies. No formal inspections of fuel storage racks are conducted beyond that resulting from normal SFP activities.

Attachment 5

To

1CAN110302

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one) ONE-TIME ACTION	TYPE (Check one) CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (If Required)
Minimum SFP temperature will be 68°F		x	
ANO-1 will comply with the requirements of 10 CFR 50.68		x	
Amend FSAR no later than next required update after approval of this TS amendment to indicate that ANO has chosen to comply with 10 CFR 50.68.	x		6 months after end of refueling outage (1R17)
Implement and maintain a coupon sampling program (Note 1)		x	

Note 1: The coupon sampling program will include the following:

- The surveillance coupons will be approximately 7" x 5" and 0.100" thick, identical in composition and manufacturing process as the Metamic in the inserts. The coupons will be mounted in stainless steel jackets simulating the actual insert design. The coupon tree will have ten or more coupons. To better simulate the conditions of the Metamic panels, the coupon tree has been designed to be installed within a flux trap. The coupon tree will be located in the Region 2 racks. The coupons will be staggered and placed adjacent to the active fuel region where, based on the burnup profile, the localized burnup is greater than the assembly average burnup. This design will maximize the dose received by the coupons as compared to the Metamic inserts and accurately simulate the flow characteristics, pool chemistry and differential metal interfaces that the Metamic inserts will experience. No welding will be used on the Metamic as per the insert design.

- The initial B_4C content of the surveillance coupons will be determined by the same chemical analysis technique used to establish the B_4C content of the Metamic in the insert. The B_4C used in the production of the Metamic is tested to determine the concentration of boron within each batch. The B_4C and aluminum matrix are accurately weighed and combined in a ratio that will assure the panels have a B_4C content higher than 24.5. Samples of the B_4C and aluminum mixture are then chemically tested to determine the B_4C concentration. Additional samples are then taken of the final product, which are also chemically tested to confirm the concentration of B_4C in Metamic. Representative samples of the batches used will undergo neutron attenuation testing to validate the B10 loading in the Metamic panels and coupons.

- Scratches will be simulated by the mechanical etching or scribing the surface of the coupon.

- The surveillance coupons will be removed and examined on a regular schedule to be established by a written procedure. Coupons will be examined on a two year basis for the first three intervals and thereafter at increasing intervals over the service life of the inserts. During the first six years, freshly discharged fuel assemblies will be placed on two sides of the coupon tree to ensure that the dose to the coupons is maximized. The initial two year period is based on providing sufficient exposure time to the environment such that any degradation would be observable.

- Measurements to be performed at each inspection will be as follows:

  A) Physical observations of the surface appearance to detect pitting, swelling or other degradation,
  B) Length, width, and thickness measurements to monitor for bulging and swelling
  C) Weight and density to monitor for material loss, and
  D) Neutron attenuation to confirm the B-10 concentration or destructive chemical testing to determine the boron content.

Enclosure 1

Holtec Report HI-2022867

Chapter 4

Revised

4.0 CRITICALITY SAFETY EVALUATION

4.1 DESIGN BASES

This section of the report documents the criticality safety evaluation for the storage of fresh and spent nuclear fuel assemblies in the ANO-1 high-density spent fuel storage racks. ANO-1 spent fuel pool currently has two regions of storage, which are currently licensed to store fuel assemblies with a maximum enrichment of 4.1 wt%.

1. Region 1 racks: These racks were originally designed with Boraflex as the poison material in a flux-trap configuration.

2. Region 2 racks: These racks are designed to store spent fuel assemblies of a specified combination of initial enrichment and discharge burnup. These racks do not currently have any poison material within the water gap between cells.

Due to the Boraflex degradation in the Region 1 racks, future credit for Boraflex is not feasible in these racks. The proposed resolution is to re-evaluate the criticality safety of the racks without credit for Boraflex and to insert poison material strips into the flux trap region of some of the Region 2 type racks. These modified Region 2 racks are identified as Region 3 racks. The new Region 3 racks will enable unrestricted fresh fuel (maximum enrichment of 5.0 wt% or nominally 4.95±0.05 wt.%) storage capability in that region. The calculations are performed with no credit for Boraflex in the Region 1 racks and with poison inserts in the Region 3 racks. All racks, including the remaining Region 2 racks were re-evaluated under the provisions of 10 CFR 50.68.

Specifically, the following evaluations were performed for ANO-1:

- The Region 1 racks were evaluated for storage of spent fuel assemblies with specific burnup requirements for the spent fuel assemblies, as a function of initial enrichments and decay times (up to 20 years). Results are summarized in Figure 4.1.1 and tabulated in Tables 4.2.1, 4.2.2 and 4.2.3.

- Fresh fuel storage in Region 1 has been assessed based in a "2-of-4" checkerboard loading with empty storage cells. (i.e., filled only with water or non-fuel bearing materials). Results are shown in Table 4.2.4 for fuel of 4.95±0.05 wt% ^{235}U enrichment.

- Region 2 racks were evaluated for storage of spent fuel assemblies with burnup requirements for the spent fuel assemblies, determined as a function of initial enrichments and decay times (up to 20 years). Results are summarized in Figure 4.1.2 and Tables 4.2.5, 4.2.6 and 4.2.7.

- Fresh fuel storage in Region 2 has been assessed based on a "2- of -4" checkerboard loading with empty storage cells. (i.e., filled only with water or non-fuel bearing materials). Results are shown in Table 4.2.8 for fuel assemblies of 4.95±0.05 wt% ^{235}U enrichment.

- The new Region 3 racks were evaluated with Metamic® panels inserted in the water gap, for storage of fresh fuel assemblies, with ^{235}U enrichments up to 4.95±0.05 wt%. Results are shown in Table 4.2.9. Region 3 may also accommodate spent fuel of any burnup for fuel assemblies up to 4.95±0.05 wt% ^{235}U enrichment.

The racks have been evaluated for Babcox & Wilcox (B&W) 15x15 spent and fresh fuel assemblies with an initial average uniform enrichment up to 5.0 wt% ^{235}U (4.95 ± 0.05 wt%). Credit is taken for poison inserts, fuel burnup, cooling time, and soluble boron in pool water as applicable per 10 CFR 50.68 and Reference 4.1.2.

The objective of this analysis is to ensure, per 10 CFR 50.68, that the racks shall remain subcritical under normal conditions with no credit for soluble boron and less than or equal to 0.95 when partial credit is taken for soluble boron in pool water, including calculation uncertainties and effects of mechanical tolerances. Reactivity effects of abnormal and accident conditions have also been evaluated to determine the required soluble boron concentration in the pool to assure that under all credible abnormal and accident conditions, the maximum reactivity will not exceed the regulatory limit of 0.95. The required soluble boron concentrations are summarized in Table 4.9.1. In this context "abnormal" refers to conditions, which may reasonably be expected to occur during the lifetime of the plant and "accident" refers to conditions, which are not expected to occur but nevertheless must be protected against. The double contingency principle [4.1.2] allows full credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time.

Applicable codes, standards, and regulations or pertinent sections thereof, include the following:

- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

- Code of Federal Regulations 10 CFR 50.68, "Criticality Accident Requirements"

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, "Spent Fuel Storage," Rev. 1 - July 1981.

- USNRC letter of April 14, 1978, to all Power Reactor Licensees – "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," including modification letter dated January 18, 1979.

- L. I. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.

- USNRC Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Rev. 2 (proposed), December 1981.

- ANSI ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors."

To assure the true reactivity will be less than the calculated reactivity, the following conservative design criteria and assumptions have been employed:

- ♦ Criticality safety analyses were based upon an infinite radial array of cells; i.e., no credit was taken for radial neutron leakage, except for evaluating accident conditions along the rack outer boundary where neutron leakage is inherent.

- ♦ Minor structural materials were neglected; i.e., spacer grids were conservatively assumed to be replaced by water.

- Because the temperature coefficient of reactivity is positive in the absence of neutron absorber panels, the analyses for Region 1 (no credit is taken in the current evaluations for the existing Boraflex poison material in these racks) and Region 2 (no poison material currently exists in these racks) type racks assumed a temperature of 150 °F. This is the design basis maximum pool water temperature. Higher temperatures would be an accident condition for which full soluble boron credit is permitted and the reactivity effects would be mitigated by the presence of the large amounts of soluble boron in the pool water.

- For Region 3 type racks, the moderator is assumed to be un-borated water at a temperature within the operating range (4 °C or 39 °F) that results in highest reactivity. Criticality calculations were performed at 20 °C (68 °F) and temperatures below 20 °C (68 °F) were assumed to be an accident condition and credit for soluble boron, as allowed by NRC guidelines, maintain a low and acceptable reactivity.

- Manufacturing tolerances of the Metamic® neutron absorber (width, thickness and B_4C loading) is included in the criticality safety evaluation. Steel components associated with the inserts are included.

- The analyses used a B&W 15x15 fuel assembly, with a maximum ^{235}U enrichment of 4.95±0.05 wt%.

- No axial blankets were assumed to be present in the fuel rods. The entire active fuel length was assumed to be of uniform enrichment.

- The calculations assumed an infinite water reflector above and below the active fuel conservatively neglecting neutron absorption in the steel of the end fittings.

- In-core depletion calculations have assumed conservative operating conditions, conservative fuel and moderator temperature, and an allowance for the average soluble boron concentrations during in-core operations.

The spent fuel storage racks are designed to accommodate the fuel assembly type listed in Table 4.1.1 with a maximum nominal initial enrichment of 4.95±0.05 wt% ^{235}U.

4.2 SUMMARY OF CRITICALITY ANALYSES

4.2.1 Normal Operating Conditions

The criticality analyses for each of the three separate regions of the spent fuel storage pool for the design basis storage conditions are summarized in Tables 4.2.1 through 4.2.9. For the fuel acceptance criteria defined in the previous section, the maximum effective neutron multiplication factor (k_{eff}) values are shown to be less than 1.0 (95% probability at the 95% confidence level) in each of the regions when no credit is taken for the presence of soluble boron in the pool. Credit for soluble boron is required to ensure k_{eff} is maintained less than 0.95. The required soluble boron concentrations are summarized in Table 4.9.1.

4.2.1.1 Region 1

The maximum k_{eff} values for storage of spent fuel were determined assuming an infinite radial array of storage cells with a finite axial length, water reflected. For each spent fuel cooling time, minimum burnup values were determined that assure the maximum k_{eff}, including calculational and manufacturing uncertainties, remains subcritical under the assumed accident condition of the loss of all soluble boron. Table 4.2.1 summarizes the results of these analyses at zero cooling time for spent fuel assemblies with an initial enrichment of 4.95 ±0.05 wt% ^{235}U. Figure 4.1.1 and Table 4.2.2 show the minimum acceptable burnup for storage of fuel assemblies of various initial enrichments and cooling times in the spent fuel. The calculated maximum reactivity includes the reactivity effect of the axial distribution in burnup and provides an additional margin for the uncertainty in the depletion calculations. The minimum soluble boron concentration required to maintain k_{eff} below 0.95, including all manufacturing tolerances and calculational tolerances, for the storage of spent fuel in the Region 1 racks, is 225 ppm.

For convenience, the minimum (limiting) burnup data shown in Table 4.2.2 may be described as a function of the nominal initial enrichment, *E*, in wt% ^{235}U by a bounding polynomial expression as shown in Table 4.2.3. Fuel assemblies with enrichments less than 2.0 wt% ^{235}U will conservatively be required to meet the burnup requirements of 2.0 wt% ^{235}U assemblies.

The maximum k_{eff}, for storage of fresh fuel assemblies of 4.95±0.05 wt% initial enrichment in the Region 1 type racks in a 2-of-4 checkerboard pattern with the alternate cells remaining empty of fuel, is 0.9443. Table 4.2.4 summarizes the results of this analysis. Based on this result, this arrangement is acceptable for storage of fresh fuel with no credit for soluble boron or for spent fuel regardless of burnup.

4.2.1.2 Region 2

The maximum k_{eff} values for storage of spent fuel were determined assuming an infinite radial array of storage cells with a finite axial length, water reflected. Calculations were performed at 0 and 20 years cooling time. For each intermediate spent fuel cooling time, minimum burnup values were determined by interpolation to assure that the maximum k_{eff}, including calculational and manufacturing uncertainties, remains subcritical under the assumed accident condition of the loss of all soluble boron. Table 4.2.5 summarizes the results of these analyses at zero cooling time for spent fuel assemblies with an initial enrichment of 4.95±0.05 wt%. Figure 4.1.2 and Table 4.2.6 show the minimum acceptable burnup for storage of fuel assemblies of various initial enrichments and cooling times in the SFP. The calculated maximum reactivity includes the reactivity effect of the axial distribution in burnup and provides an additional margin for the uncertainty in the depletion calculations. The minimum soluble boron concentration required to maintain k_{eff} below 0.95, including all manufacturing and calculational tolerances, for the storage of spent fuel allowed in the Region 2 racks is 180 ppm.

For convenience, the minimum (limiting) burnup data shown in Table 4.2.6 may be described as a function of the nominal initial enrichment, *E*, in wt% ^{235}U by a bounding polynomial expression as shown in Table 4.2.7. Fuel assemblies with enrichments less than 2.0 wt% ^{235}U will conservatively be required to meet the burnup requirements of 2.0 wt% ^{235}U assemblies.

The maximum k_{eff}, for storage of fresh unburned fuel assemblies of enrichment 4.95 ± 0.05 wt% in the Region 2 in a 2-of-4 checkerboard pattern with the alternate cells remaining empty of fuel, is 0.9621. Table 4.2.8 summarizes the results of this analysis and confirms that this arrangement is acceptable for storage of fresh fuel or spent fuel regardless of burnup, without requiring any credit for soluble boron. The minimum soluble boron concentration required to maintain k_{eff} below the 0.95 limit, including all manufacturing and calculational tolerances, for the storage of fresh fuel assemblies in this configuration in the Region 2 racks is 100 ppm.

4.2.1.3 Region 3

The Region 3 racks were analyzed for the storage of 4.95±0.05 wt% fresh fuel assemblies. The maximum k_{eff}, for storage of fresh fuel assemblies in the Region 3 racks is 0.9903. Table 4.2.9 summarizes the results of this analysis, and confirms that this arrangement is acceptable for storage of fresh unburned fuel or spent fuel regardless of burnup. The minimum soluble boron concentration required in Region 3 to maintain k_{eff} below 0.95, including all manufacturing and calculational tolerances, is 400 ppm.

4.3 REFERENCE DESIGN INPUT DATA

4.3.1 Reference Fuel Assembly

The spent fuel storage racks are designed to accommodate B&W 15x15 fuel assemblies. The design specifications for the B&W fuel assemblies designs, as used for this analysis, are given in Table 4.1.1.

4.3.2 Region 1 Fuel Storage Cells

Figure 4.3.1 shows the calculational model of the nominal Region 1 spent fuel storage cell. The Region 1 storage cells are composed of stainless steel boxes separated by a gap. The 0.075 ± 0.004 thick steel walls define the storage cells, which have an 8.970 -.05/+.025 inch inside dimension. A 0.020±0.004 inch stainless steel sheath is around the gap and defines the boundary of the flux-trap water-gap used to augment reactivity control. The cells are located on a lattice spacing of 10.65 inches in both directions. Stainless steel channels connect the storage cells in a rigid structure and define the flux-trap of 1.31 inches, between the sheathing of adjacent cells.

4.3.3 Region 2 Fuel Storage Cells

Figure 4.3.2 shows the calculational model of the nominal Region 2 spent fuel storage cell. The Region 2 storage cells area also have a flux trap between adjacent cells and are composed of stainless steel boxes separated by a gap. The straight portion of the flux trap is 7.5±0.06 inches. The measured flux trap water gap of 1.456 +0.12/-0.08 inches was used in the analysis. The 0.062 ±0.004 thick steel walls define the storage cells, which have an 8.97 +0.050/-0.025 inch nominal inside dimension. The measured value of the flux trap water gap corresponded to a Box inside dimension

(ID) of 9.07 +0.50/-0.025 inches because of the bow in the cell walls. This value of the Box ID was used in the analysis. The cells are located on a lattice spacing of 10.65 inches in both directions. No additional water gaps exist between adjacent Region 2 cells in a rack.

4.3.4 Region 3 Fuel Storage Cells

The Region 3 storage cells are identical to Region 2 storage cells except that the Metamic® Poison panels will be inserted into the flux trap gaps. The poison panels are designed to be 7.00 ±0.0625 inches wide with a minimum Boron Carbide (B_4C) content of 25±0.5 weight percent. The criticality safety analysis conservatively assumes that the poison panels are 6.7 inches wide. These Metamic® panels are supported by stainless steel and held by appropriate mechanisms to remain close to the flux trap walls. The stainless steel support consists of a plate 0.018 inch thick on the outside of the Metamic panels and a series of straps internally, with an equivalent (smeared) thickness of 0.0085 inches on each Metamic panel.

4.4 ANALYTICAL METHODOLOGY

4.4.1 Reference Design Calculations

The principal methods for the criticality analyses of the storage racks include the following codes: (1) MCNP4a [4.4.1], (2) KENO Va [4.4.2], and (3) CASMO-4 [4.4.5-4.4.7]. MCNP4a is a continuous energy three-dimensional Monte Carlo code developed at the Los Alamos National Laboratory. KENO Va is a three-dimensional multigroup Monte Carlo code developed at the Oak Ridge National Laboratory as part of the SCALE 4.3 package [4.4.3]. The KENO Va calculations used the 238-group SCALE cross-section library and NITAWL [4.4.4] for ^{238}U resonance shielding effects (Nordheim integral treatment). Benchmark calculations, presented in Appendix 4A, indicate a bias of 0.0009 with an uncertainty of ± 0.0011 for MCNP4a and 0.0030 ± 0.0012 for KENO Va, both evaluated with the 95% probability at the 95% confidence level [4.1.1].

Fuel depletion analyses during core operation were performed with CASMO-4, two-dimensional multigroup transport theory codes based on transmission probabilities [4.4.5 - 4.4.7]. Restarting the CASMO-4 calculations in the storage rack geometry yields the two-dimensional infinite multiplication factor (k_∞) for the storage rack. CASMO-4 was also used to determine the reactivity uncertainties (differential calculations) of manufacturing tolerances and the reactivity effects of variations in the water temperature and density.

In the geometric models used for the calculations, each fuel rod and its cladding were described explicitly and reflecting boundary conditions were used in the radial direction, which has the effect of creating an infinite radial array of storage cells. Monte Carlo calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the MCNP4a and KENO Va calculated reactivity and to assure convergence, a minimum of 1 million neutron histories were accumulated in each calculation. Three-dimensional MCNP calculations were necessary to describe the geometry of the checkerboard cases. However, MCNP cannot perform depletion calculations, and depletion calculations were performed with CASMO-4. Explicit description of the fission product nuclide concentrations in the spent fuel was determined from the CASMO-4 calculations and used in the MCNP calculations. To compensate for those few fission product nuclides that are not in the MCNP library, an equivalent boron-10 concentration in the fuel was determined which produced the same reactivity in MCNP as the CASMO-4 result. This methodology explicitly incorporates approximately 40 of the most important fission products, accounting for all but about 1% in reactivity. The remaining ~1 % in reactivity is included by the equivalent B-10 concentration in the fuel.

4.4.2 Fuel Burnup Calculations and Uncertainties

CASMO-4 was used for burnup calculations in the hot operating condition. Conservatively bounding moderator and fuel temperatures and the average operating soluble boron concentrations (900 ppm) were used to assure the high plutonium production and hence conservatively high values of reactivity. Since critical experiment data with spent fuel is not available for determining the uncertainty in depletion calculations, an allowance for uncertainty in reactivity was assigned based upon other considerations [4.1.2]. Assuming the uncertainty in depletion calculations is less than 5% of the total reactivity decrement; a burnup dependent uncertainty in reactivity for burnup calculations was assigned. Thus, the burnup uncertainty varies (increases) with burnup. This allowance for burnup uncertainty was included in determination of the acceptable burnup versus enrichment combinations.

4.4.3 Effect of Axial Burnup Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower regions. At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of high neutron leakage. Consequently, it is expected that over most of the burnup history, fuel assemblies with distributed burnups will exhibit a slightly lower reactivity than that calculated for the uniform average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup.

Among others, Turner [4.4.8] has provided generic analytic results of the axial burnup effect based upon calculated and measured axial burnup distributions. These analyses confirm the minor and generally negative reactivity effect of the axially distributed burnups at values less than about 30 GWD/MTU with small positive reactivity effects at higher burnup values. Calculations were performed based upon a burnup distribution provided by ANO. These calculations were performed in MCNP4a with 10 zone axial calculations, using specific (CASMO-4) concentrations of actinides and fission product nuclides in each zone. Results of these calculations, therefore, inherently include the effect of the axial distribution in burnup.

4.4.4 BPRAs and APSRs

The fuel assemblies used at the ANO-1 Nuclear Power Plant contain poison rods such as the Burnable Poison Rods Assemblies (BPRAs) and Axial Power Shaping Rods (APSRs). These rods are inserted into the guide tubes during the in-core depletion process and are stored in fuel assemblies in the rack cells. Analyses show that the fuel assemblies, which contained the BPRAs during depletion in the core, are more reactive in the racks than those that contained APSRs or no burnable absorber in the burnup range of interest in this analysis. CASMO-4 calculations performed to obtain the isotopic compositions of the spent fuel assemblies, which were subsequently used in the MCNP calculations, were performed with BPRAs in the fuel assemblies during core operation.

4.4.5 MCNP4a Temperature Correction

The reactivity for non-poisoned racks in the spent fuel pool increases with pool water temperature. The maximum bulk pool water temperature is 150 °F. However, since the Doppler treatment and cross-sections in MCNP4a are valid only at 20 °C (68 °F), the delta-k determined in CASMO-4 from 20 °C (68 °F) to the limiting temperature described in section 4.8.1 is included in the final k_{eff} calculation.

4.4.6 Long-Term Changes in Reactivity

At reactor shutdown, the reactivity of the fuel initially decreases due to the growth of Xe-135. Subsequently, the Xenon decays and the reactivity increase to a maximum at about a hundred hours when the Xenon is gone. Therefore, for conservatism, the Xe is set to zero in the calculations to assure maximum reactivity. During the next 50 years, the reactivity continuously decreases due primarily to ^{241}Pu decay and ^{241}Am growth. Credit for this decay and for changes in fission product concentrations is included in calculations of the decrease in reactivity in long-term storage (up to 20 years). The CASMO-4 code includes the capability of tracking the decay of the actinides and the most significant product nuclides during long-term storage.

4.5 REGION 1 CRITICALITY ANALYSES AND TOLERANCES

4.5.1 Nominal Design Case

For the nominal storage cell design in Region 1, the criticality safety analyses for the two different storage patterns are summarized in Tables 4.2.1, 4.2.2 and 4.2.4. This data confirms that the maximum reactivity in Region 1 remains subcritical (less than the regulatory limit $k_{eff} < 1.0$) under the assumed condition of the loss of all soluble boron in the pool water. Figure 4.1.1 shows the limiting burnup values for fuel of other enrichments and cooling times (see also Table 4.2.2).

4.5.2 Uncertainties Due to Tolerances

The reactivity effects of manufacturing tolerances are summarized in Tables 4.2.1 and 4.2.4. All of the individual reactivity allowances were separately calculated for the reference fuel assembly and a statistical combination of uncertainties was used. The tolerances include the fuel storage area I.D., rack material thickness, water gap, fuel enrichment and density.

4.5.3 Eccentric Fuel Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell. However, calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). These calculations indicated that the reactivity effect is slightly positive. Therefore, the uncertainty for eccentricity is included in the calculations for the final k_{eff} in Tables 4.2.1 and 4.2.4.

4.6 REGION 2 CRITICALITY ANALYSES AND TOLERANCES

4.6.1 Nominal Design Case

For the nominal storage cell design in Region 2, the criticality safety analyses are summarized in Tables 4.2.5, 4.2.6 and 4.2.8. This data confirms that the maximum reactivity in Region 2 remains subcritical (less than the regulatory limit $k_{eff} < 1.0$) under the assumed condition of the loss of all soluble boron in the pool water. Figure 4.1.2 (and Table 4.2.6) summarizes the limiting fuel burnups for fuel assemblies of other enrichments and cooling times.

4.6.2 Uncertainties Due to Tolerances

The reactivity effects of manufacturing tolerances are summarized in Tables 4.2.5 and 4.2.8. All of the individual reactivity allowances were separately calculated for the reference fuel assembly and a statistical combination of uncertainties was used. The tolerances include the fuel storage area I.D., rack material thickness, water gap, fuel enrichment and density.

4.6.3 Eccentric Fuel Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell. However, calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). These calculations indicate that the reactivity effect is slightly positive. Therefore, the uncertainty for eccentricity is included in the calculations for the final k_{eff} in Tables 4.2.5 and 4.2.8.

4.7 REGION 3 CRITICALITY ANALYSES AND TOLERANCES

4.7.1 Nominal Design Case

For the nominal storage cell design in Region 3, the criticality safety analyses are summarized in Table 4.2.9. This data confirms that the maximum reactivity in Region 3 remains subcritical (less than the regulatory limit $k_{eff} < 1.0$) under the assumed condition of the loss of all soluble boron in the pool water. The minimum design basis temperature is 68 °F (20 °C) and any lower temperature is considered an accident condition for which credit for the soluble boron in the pool water is assumed. Administrative procedures will be used and preclude a temperature below 20 °C during normal storage.

4.7.2 Uncertainties Due to Tolerances

The reactivity effects of manufacturing tolerances are tabulated in Table 4.2.9. All of the individual reactivity allowances were separately calculated for the reference fuel assembly and a statistical combination of uncertainties was used. The tolerances include the fuel storage area I.D., rack material tolerances (including thickness and B-10 loading tolerance for the Metamic plates), water gap, fuel enrichment and density.

4.7.3 Eccentric Fuel Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell. However, calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). These calculations indicate that the reactivity effect is slightly negative, so it is conservatively not included.

4.8 ABNORMAL AND ACCIDENT CONDITIONS IN THE SPENT FUEL POOL RACKS

4.8.1 Temperature and Water Density Effects

The moderator temperature coefficient of reactivity in both Region 1 and Region 2 is positive. A moderator temperature of 20 °C (68 °F) was assumed for the reference MCNP4a calculations and the increase in reactivity to the maximum permissible bulk pool water temperature of 150 °F is included (CASMO-4 calculation) as an additive term in the calculation of the maximum k_{eff}. This assures that the true reactivity will always be lower over the expected range of water temperatures. The reactivity effects of the pool water temperature effects on reactivity have been evaluated using CASMO-4.

The moderator temperature coefficient of reactivity for the Region 3 is negative. The minimum design basis temperature is 20°C (68°F) and any lower temperature is considered an accident condition for which credit for the soluble boron in the pool water is assumed. Administrative procedures will be used to preclude a temperature below 20 °C during normal storage.

4.8.2 Lateral Rack Movement

Lateral motion of the storage racks under postulated seismic conditions could potentially alter the spacing between racks. Under these conditions, credit for the soluble boron (permitted under accident conditions) would maintain the k_{eff} at a value well below the maximum allowable. The double contingency principle requires the consideration of only one accident at one time. Nevertheless, the separation (water-gap) between rack modules is sufficiently large that even for the maximum movement expected under seismic excitation, the water gap remains larger than the water gap within the Region 1 modules. In the Region 2 and the Region 3 racks, the k_{eff} is independent of the inter-module water gap and is not sensitive to any potential seismic induced movement of the modules. The water-gap structure in each cell is included in the analysis and precludes any closer proximity between modules.

4.8.3 Abnormal Location of a Fuel Assembly

The misplacement of a fresh unburned fuel assembly of the highest permissible reactivity would, in the absence of soluble poison, result in exceeding the regulatory limit (k_{eff} < 1.0). This could occur if a fresh fuel assembly of the highest permissible initial enrichment (4.95±0.05 wt% ^{235}U) were to be inadvertently loaded into a Region 1 or Region 2 storage cell, which are intended to store spent fuel assemblies or remain empty. Calculations confirmed that the highest reactivity, including uncertainties, for the worst case postulated fuel mis-loading accident condition (fresh fuel assembly in

Region 2 cell intended to remain empty) would exceed the limit on reactivity in the absence of soluble boron. Soluble boron in the spent fuel pool water, for which credit is permitted under these accident conditions, would assure that the reactivity is maintained substantially less than the design limitation. Calculations indicate that a soluble boron concentration of 800 ppm is adequate to assure that the maximum k_{eff} does not exceed 0.95. The soluble boron concentration in Region 1 is bounded by the soluble boron concentration for the Region 2 accident condition.

In addition, the mislocation of a fresh unburned fuel assembly could occur if a fresh fuel assembly of the highest permissible initial enrichment (4.95 ±0.05 wt%) were to be accidentally mislocated outside of a Region 1 or Region 2 storage rack, with the rack fully loaded. However, this is an area of high neutron leakage and the reactivity effect would be bounded by that of a fuel assembly accidentally misloaded internal to Region 1 or Region 2 storage module. The gaps between the racks are sufficiently small to preclude the accidental mislocation of a fuel assembly in between storage rack modules.

4.8.4 Dropped Fuel Assembly

For the case in which a fuel assembly is assumed to be dropped horizontally on top of a rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel region sufficient to preclude neutron coupling (with fuel in the storage rack). Consequently, the horizontal fuel assembly drop accident will not result in a significant increase in reactivity. Furthermore, the soluble boron in the spent fuel pool water assures that the true reactivity is always less than the limiting value for this dropped fuel accident.

Analyses were performed to evaluate the potential loss of Metamic® poison panels in the Region 3 racks by means of a dropped fuel assembly. A very conservative bounding accident condition was analyzed postulating the loss of all Metamic® absorber material throughout the entire Region 3 storage racks. For this accident analysis, the Tech Spec limit of 1600 ppm soluble boron was assumed. The results of this postulated accident condition show that the maximum k_{eff} is well below 0.95, including bias and tolerance uncertainties. This is a very conservative evaluation since an actual dropped assembly at most, would be expected to only damage a maximum of eight Metamic® panels (4 inserts).

It is also possible to vertically drop an assembly into a location occupied by another assembly. Such a vertical impact, would, at most cause a small compression of the stored assembly, reducing the water-to-fuel ratio and thereby reducing reactivity. In addition the distance between the active fuel regions of both assemblies will be more than sufficient to ensure no neutron interaction between the two assemblies.

Dropping of an assembly into an unoccupied cell could result in a localized deformation of the baseplate of the rack. The immediate eight surrounding fuel cells could also be affected. However, the amount of deformation for these cells would be considerably less. The resultant effect would be the lowering of a few fuel assemblies in the area near the deformation. Since the Metamic inserts in Region 3 are designed to sit on the baseplate, the localized deformation of the baseplate was considered. The Metamic® poison panel inserts in the Region 3 type racks are designed to sit on the baseplate and could potentially move downward and uncover a portion of the active fuel. The resulting geometry is bound by the previously discussed configuration in which a complete loss of all Metamic® inserts was assumed with no change in the active fuel region alignment. Therefore, the presence of the boron concentration assures the maximum k_{eff} is well below the 0.95 acceptance criteria.

4.9 SOLUBLE BORON DILUTION EVALUATION

The soluble boron in the spent fuel pool water is normally a minimum of 1600 ppm under operating conditions. Significant loss or dilution of the soluble boron concentration is extremely unlikely, if not incredible. Nonetheless, an evaluation was performed based on the ANO spent fuel pool data. The minimum required soluble boron concentration in the spent fuel pool water for various conditions are summarized in Table 4.9.1.

The required minimum soluble boron concentration is 400 ppm under normal conditions and 800 ppm for the most serious credible accident scenario. The volume of water in the pool is 256,860 gallons. Large amounts of unborated water would be necessary to reduce the boron concentration from 1600 ppm to 800 ppm or 400 ppm. Abnormal or accident conditions are discussed below for either low dilution rates (abnormal conditions) or high dilution rates (accident conditions). It should be noted that routine surveillances to measure the soluble boron concentrations in the pool water are required by Technical Specifications at least weekly.

Small dilution flow around pump seals and valve stems or mis-aligned valves could possibly occur in the normal soluble boron control system or related systems. Such failures might not be immediately detected. These flow rates would be of the order of 2 gpm (maximum) and the increased frequency of makeup flow might not be observed. However, an assumed loss flow-rate of 2 gpm dilutions flow rate would require some 123 days to reduce the boron concentration to the minimum required 400 ppm required under normal conditions or 62 days to reach the 800 ppm required for the most severe fuel handling accident. Routine surveillance measurements of the soluble boron concentration would readily detect the reduction in soluble boron concentration with ample time for corrective action.

Under certain accident conditions, it is conceivable that a high flow rate of unborated water could flow onto the top of the pool. Such an accident scenario could result from rupture of a demineralized water supply line or possibly the rupture of a fire protection system header, both events potentially allowing unborated water to spray onto the pool. A flow rate of up to 2500 gpm could possibly flow onto the spent fuel pool as a result of a rupture of the fire protection line. This would be the most serious condition and bounds all other accident scenarios. Conservatively assuming that all the unborated water from the break poured onto the top of the pool and further assuming instantaneous mixing of the unborated water with the pool water, it would take approximately 142 minutes to dilute the soluble boron concentration to 400 ppm, which is the minimum required concentration to maintain k_{eff} below 0.95 under normally operating conditions. In this dilution accident, some 355,000 gallons of water would spill on the auxiliary building floor. Well before the spilling of such a large volume of water, multiple alarms would have alerted the control room of the accident consequences (including the fuel pool high-level alarm, the fire protection system pump operation alarm, and the floor drain receiving tank high level alarm). For this high flow rate condition, 71 minutes would be required to reach the 800 ppm required for the most severe fuel handling accident.

Instantaneous mixing of pool water with the water from the rupture of the demineralized water supply line is an extremely conservative assumption. Water falling on to the pool surface would mix with the top layers of pool water and the portions of the mixed volumes would continuously spill out of the pool. The density difference between water at 150 °F (maximum permissible pool bulk water temperature) and at the temperature of the demineralizer water supply is small. This density difference will not cause the water falling on to the pool surface to instantaneously sink down into the racks overcoming the principal driving force for the flow in the pool, which is the buoyancy force generated in the spent fuel pool racks region due to the heat generation from the spent fuel in the racks. This would further enhance the mixing process between the pool water and spilled water above the racks.

For the fire water system line break, upon the initial break, the fire protection system header pressure would drop to the auto start setpoint of the fire protection pumps. The start is accompanied with an alarm in the main control room. The annunciator response is to dispatch an operator to find the source of the pump start. Approximately 5 minutes into the event, a Spent Fuel Pool high level alarm would be received in the main control room, assuming that the Spent Fuel Pool level started at the low alarm. The annunciator response for high Spent Fuel Pool level is to investigate the cause. The coincidence of the 2 alarms would quickly lead to the discovery of the failure of the fire protection system and sufficient time to isolate the failure.

The maximum flow rate for a failure of the demineralized water header would provide approximately 900 gpm into the Spent Fuel Pool. Failure of the demineralized water header is not accompanied with an alarm; however, the time to dilute the Spent Fuel Pool from 1600 to 400 ppm is greater than the bounding case described above. In this scenario, there is sufficient time to isolate the failure and to prevent the spilling of some 355,000 gallons of water.

The analysis assume that for a double-ended break in the fire water system piping, the stream of water will arch through the air some 40 feet falling on top of the pool. This is virtually an incredible event. Should the stream of water fall upon the pool deck, a 3 inch high curb would channel some of the water to the pool drain and prevent all of the water from reaching the pool. Furthermore, the evaluation also assumes at least 3 independent and concurrent accidents occur simultaneously:

- Large amount of water flowing from the double-ended pipe break would remain un-detected and is ignored.
- Pool water high level alarms either fail or are ignored.
- Alarms indicating large amounts of water flowing into the floor drain have failed or are ignored.

Considering all related facts, a significant dilution of the pool soluble boron concentration in a short period of time without corrective action is not considered a credible event.

It is not considered credible that multiple alarms would fail or be ignored or that the spilling of large volumes of water would not be observed. Therefore, such a major failure would be detected in sufficient time for corrective action to avoid violation of an administrative guideline and to assure that the health and safety of the public is protected.

4.10 NEW FUEL STORAGE RACKS CRITICALITY ANALYSIS

The New Fuel Storage Vault is intended for the receipt and storage of fresh fuel under normally dry conditions where the reactivity is very low. To assure criticality safety under accident conditions and to conform to the requirements of General Design Criterion 62, "Prevention of Criticality in Fuel

Storage and Handling", two separate criteria must be satisfied as defined in NUREG-0800, Standard Review Plan 9.1.1, "New Fuel Storage". These criteria are as follows:

- When fully loaded with fuel of the highest anticipated reactivity and flooded with clean unborated water, the maximum reactivity, including uncertainties, shall not exceed a k_{eff} of 0.95.

- With fuel of the highest anticipated reactivity in place and assuming the optimum hypothetical low density moderation, (i.e., fog or foam), the maximum reactivity shall not exceed a k_{eff} of 0.98.

The New Fuel Storage Vault provides two 4 x 9 storage rack modules with cell array of storage locations arranged on a 21.00 inch lattice spacing. Calculations were made with the 238-group NITAWL KENO Va code package (SCALE 4.3), a three-dimensional Monte Carlo analytical technique, with fresh fuel assemblies with 4.95 wt% initial enrichment. These calculations were made for various moderator densities and the results shown in Figure 4.10.1, shows that the peak reactivity (optimum moderation) occurs at 9% moderator density. The calculations for the configuration illustrated in Figure 4.10.2 confirm that five locations in each of the storage racks are required to remain empty in order to meet the regulatory limits. Results of the criticality safety analyses are summarized in Table 4.10.1 for the two accident conditions for fuel assemblies of 4.95±0.05 wt% initial enrichment. The maximum reactivity at 9% moderator density is 0.9708, including uncertainties, which is within the regulatory limit of 0.98, thus confirming the acceptability of the New Fuel Vault for 4.95±0.05 wt% fuel.

Additional calculations at 9% moderator density, performed for the storage pattern depicted in Figure 4.10.3, show that this storage configuration is acceptable for storage of fresh fuel assemblies of up to 4.20 wt% enrichment with four locations in each rack array required to remain empty.

In the flooded condition (clean unborated water), the storage locations are essentially isolated from each other (neutronically). Under these conditions and with fuel of 4.95±0.05 wt% enrichment, the maximum reactivity, including all known uncertainties is less than the regulatory limit of 0.95 for k_{eff}, thus confirming the acceptability of the NFV for 4.95 wt% fuel in the fully flooded accident condition. At 4.2 wt% enrichment in the flooded condition, the reactivity will be substantially lower than that for 4.95±0.05 wt% enrichment and would therefore be acceptable for storage. (Note: in the fully flooded conditions the fuel assemblies are neutronically isolated from each other and the reactivity is independent of the presence or number of blanked-off cells.)

4.11 FUEL HANDLING EQUIPMENT

Criticality safety evaluations were also performed for handling of fresh fuel assemblies during transfer from the new fuel vault to the reactor core, including the new-fuel elevator, the up-ender and fuel carriage, and the temporary storage rack within the transfer canal. The new fuel elevator is located on the south wall of the pool facing the Region 1 spent fuel storage racks. This device can position a fresh fuel assembly 16 inches (assembly center line) from the wall. The distance from the wall to the edge of the rack is 24.5 inches. A distance of 7.845 inches exists between the centerline of the assembly in the elevator and the edge of the closest fuel storage cell in the rack. For normal operation, the maximum reactivity with fuel in the New Fuel Elevator (with Region 1 loaded) is 0.9512 in unborated water. This is reduced to 0.9271 with credit for 100 ppm soluble boron. Additional calculations were performed to evaluate the effect of accidentally dropping or misplacing an assembly adjacent to the New Fuel Elevator while it is loaded. A most reactive location for the dropped assembly was determined. A credit of 700 ppm boron will ensure k-effective remains below 0.95 should such an event occur. The New Fuel Elevator therefore meets the criticality acceptance criteria defined in 10 CFR 50.68. The Upender/ Fuel Carriage device handles a single assembly. The maximum reactivity of a single fresh assembly containing 4.95 w/o + 0.05 enriched fuel in water is 0.9425. Furthermore for a postulated accident in which a second fresh assembly was positioned near the Upender/Fuel Carriage, the presence of soluble boron (1600 ppm minimum) excludes the possibility of any criticality concern.

The transfer canal incorporates a 7-cell temporary storage rack on a linear array at a 21-$^{1}/_{8}$ inch spacing (6 locations for fuel assemblies and 1 location for a damaged fuel). The maximum k-effective for normal operation of this rack was determined to be 0.9413. Evaluations of a potential mis-placement of a fresh fuel assembly at a position of closest approach to another assembly in the spent fuel rack, separated only by the structure of the temporary rack, shows that the maximum k_{eff} (in the absence of any soluble boron) would be 0.9698. The presence of 200 ppm soluble boron would be sufficient to maintain the maximum k-effective below 0.95. However, the transfer canal, during operations, would always contain the minimum 1600 ppm boron (or usually more), significantly reducing reactivity and further eliminating any criticality concern.

The fuel upender and fuel carriage use a single fuel assembly. The maximum reactivity of a single fresh fuel assembly in both these devices is well below the regulatory limit of 0.95. Thus, there are no criticality safety concerns with the upender and the fuel carriage. The accidental placement of a second fresh fuel assembly or the drop of an assembly adjacent to the shroud in the upender is a credible event but does not result in any neutronic interaction due to the distance between the assemblies. Furthermore, for such a postulated accident condition, presence of soluble boron in the pool water (minimum of 1600 ppm) would exclude the possibility of criticality concern.

4.12 REFERENCES FOR SECTION 4.0

[4.1.1] M. G. Natrella, Experimental Statistics, National Bureau of Standards Handbook 91, August 1963.

[4.1.2] L.I. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.

[4.4.1] J.F. Briesmeister, Editor, "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A," LA-12625, Los Alamos National Laboratory (1993).

[4.4.2] L.M. Petrie and N.F. Landers, "KENO Va - An Improved Monte Carlo Criticality Program with Supergrouping," Volume 2, Section F11 from "SCALE: A Modular System for Performing Standardized Computer Analysis for Licensing Evaluation" NUREG/CR-0200, Rev. 4, January 1990.

[4.4.3] "SCALE 4.3: A Modular System for Performing Standardized Computer Analysis for Licensing Evaluation For Workstations and Personal Computers, Volume 0," CCC-545, ORNL-RSICC, Oak Ridge National Laboratory (1995).

[4.4.4] N.M. Greene, L.M. Petrie and R.M. Westfall, "NITAWL-II: Scale System Module for Performing Shielding and Working Library Production," Volume 1, Section F1 from "SCALE: A Modular System for Performing Standardized Computer Analysis for Licensing Evaluation" NUREG/CR-0200, Rev. 4, January 1990.

[4.4.5] M. Edenius, K. Ekberg, B.H. Forssén, and D. Knott, "CASMO-4 A Fuel Assembly Burnup Program User's Manual," Studsvik/SOA-95/1, Studsvik of America, Inc. and Studsvik Core Analysis AB (proprietary).

[4.4.6] D. Knott, "CASMO-4 Benchmark Against Critical Experiments", SOA-94/13, Studsvik of America, Inc., (proprietary).

[4.4.7] D. Knott, "CASMO-4 Benchmark Against MCNP," SOA-94/12, Studsvik of America, Inc., (proprietary).

[4.4.8] S.E. Turner, "Uncertainty Analysis - Burnup Distributions", presented at the DOE/SANDIA Technical Meeting on Fuel Burnup Credit, Special Session, ANS/ENS Conference, Washington, D.C., November 2, 1988.

Table 4.1.1

Fuel Assembly Specifications

Assembly Data	
Rod Array Size	15x15
Rod Pitch (inches)	0.5680
Active Fuel Length (inches)	144
Fuel Rod Data	
Total Number of Fueled Rods	208
Fuel Rod Outer Diameter (inches)	0.430
Fuel Rod Inner Diameter (inches)	0.377
Cladding Thickness (inches)	0.0265
Cladding Material	Zircaloy
Pellet Diameter (inches)	0.370
UO_2 Stack Density, gms/cc	10.412±0.20
Guide Tube Data	
Number of Tubes	16
Outer Diameter (inches)	0.530
Wall Thickness (inches)	0.016
Material	Zircaloy
Instrument Tube Data	
Number of Tubes	1
Outer Diameter (inches)	0.493
Wall Thickness (inches)	0.026
Material	Zircaloy

Table 4.2.1

Summary of the Criticality Safety Analyses for
Storage of Spent Fuel Assemblies in Region 1 Racks

Reference k_{eff}	0.9710
Burnup, MWD/KgU	35.4
MCNP4a Bias	0.0009
Temperature Bias	0.0100
Uncertainty in MCNP4a Bias	±0.0011
MCNP4a Statistical (95/95) Uncertainty	±0.0007
Manufacturing Tolerance Uncertainty	±0.0041
Enrichment Tolerance Uncertainty	±0.0027
Depletion Uncertainty	±0.0119
Fuel Eccentric Positioning Uncertainty	±0.0020
Statistical Combination of Uncertainties	±0.0131
Maximum k_{eff}	0.9950
Regulatory Limiting k_{eff}	1.0000

Table 4.2.2

Minimum Burnup Required for Storage of Spent Fuel Assemblies in the Region 1 Racks

BURNUP, MWD/KgU					
Average Enrichment, wt% ^{235}U	**0 Years Cooling Time**	**5 Years Cooling Time**	**10 Years Cooling Time**	**15 Years Cooling Time**	**20 Years Cooling Time**
2	0.50	0.375	0.25	0.125	0.00
2.5	7.50	7.35	7.20	7.05	6.90
3.0	13.60	13.10	12.60	12.10	11.60
3.5	18.70	18.03	17.35	16.68	16.00
4.0	24.27	23.45	22.64	21.82	21.00
4.5	30.00	29.05	28.10	27.15	26.20
4.95	35.40	34.08	32.75	31.43	30.10

Table 4.2.3

Bounding Polynomial Fits to Determine Minimum Acceptable Bumup for Storage of Spent Fuel Assemblies Storage in Region 1 Racks as a Function of Initial Average Enrichment

Decay Time, Years	Burnup, MWD/KgU
0	$-0.1661*E^4+2.791*E^3-17.013*E^2+55.795*E-62.57$
5	$-0.3014*E^4+4.6636*E^3-26.447*E^2+75.84*E-77.88$
10	$-0.443*E^4+6.6239*E^3-36.324*E^2+96.837*E-93.94$
15	$-0.5782*E^4+8.4965*E^3-45.758*E^2+116.88*E-109.27$
20	$-0.7198*E^4+10.457*E^3-55.635*E^2+137.88*E-125.30$

Note: E = Initial average enrichment in wt% ^{235}U

Table 4.2.4

Summary of the Criticality Safety Analyses for a 2-of-4 Checkerboard Storage of Fresh Fuel Assemblies and Empty Cells in Region 1 Racks

Reference k_{eff}	0.9264
MCNP4a Bias	0.0009
Temperature Effect	0.0117
Axial Burnup Distribution Penalty	Not Applicable
Uncertainty in MCNP4a Bias	±0.0011
MCNP4a Statistics (95/95) Uncertainty	±0.0007
Manufacturing Tolerance Uncertainty	±0.0044
Enrichment Tolerance Uncertainty	±0.0019
Depletion Uncertainty	Not Applicable
Fuel Eccentric Positioning Uncertainty	±0.0017
Statistical Combination of Uncertainties	±0.0053
Maximum k_{eff}	0.9443
Regulatory Limiting k_{eff}	1.0000

Table 4.2.5

Summary of the Criticality Safety Analyses for Storage of Spent Fuel Assemblies in Region 2 Racks

Initial Enrichment, wt% ^{235}U	4.95±0.05
Burnup, MWD/KgU	41.00
Cooling Time, years	0
Reference k_{eff}	0.9619
MCNP4a Bias	0.0009
Temperature Effect	0.0112
Uncertainty in MCNP4a Bias	±0.0011
MCNP4a Statistics (95/95) Uncertainty	±0.0007
Manufacturing Tolerance Uncertainty	±0.0094
Enrichment Tolerance Uncertainty	±0.0028
Depletion Uncertainty	±0.0133
Fuel Eccentric Positioning Uncertainty	±0.0088
Statistical Combination of Uncertainties	±0.0188
Maximum k_{eff}	0.9928
Regulatory Limiting k_{eff}	1.0000

Table 4.2.6

Minimum Burnup Required for Storage of Spent Fuel Assemblies in the Region 2 Racks

BURNUP, MWD/KgU					
Average Enrichment, wt% ^{235}U	**0 Years Cooling Time**	**5 Years Cooling Time**	**10 Years Cooling Time**	**15 Years Cooling Time**	**20 Years Cooling Time**
2.00	4.00	4.00	4.00	4.00	4.00
2.50	11.50	11.00	10.50	10.00	9.50
3.00	17.80	17.10	16.40	15.70	15.00
3.50	23.50	22.63	21.75	20.88	20.00
4.00	29.20	28.15	27.10	26.05	25.00
4.50	35.00	33.63	32.25	30.88	29.50
4.95	41.00	39.25	37.50	35.75	34.00

Table 4.2.7

Bounding Polynomial Fits to Determine Minimum Acceptable Burnup for Storage of Spent Fuel Assemblies Storage in Region 2 Racks as a Function of Initial Enrichment

Decay Time, Years	Burnup, MWD/KgU
0	$0.6257E^3 - 6.8758E^2 + 36.293E - 46.0$
5	$0.4934*E^3 - 5.5023*E^2 + 31.23*E - 40.3$
10	$0.3634*E^3 - 4.1511*E^2 + 26.244*E - 34.7$
15	$0.2311*E^3 - 2.776*E^2 + 21.186*E - 29.0$
20	$0.1011*E^3 - 1.4265*E^2 + 16.19*E - 23.4$

Note: E = Initial average enrichment in wt% ^{235}U

Table 4.2.8

Summary of the Criticality Safety Analyses for a 2-of-4 Checkerboard Storage of Fresh Fuel Assemblies and Empty Cells in Region 2 Racks

Reference k_{eff}	0.9359
MCNP4a Bias	0.0009
Temperature Effect	0.0129
Uncertainty in MCNP4a Bias	±0.0011
MCNP4a Statistics (95/95) Uncertainty	±0.0007
Manufacturing Tolerance Uncertainty	±0.0119
Enrichment Tolerance Uncertainty	±0.0020
Depletion Uncertainty	Not Applicable
Fuel Eccentricity Positioning Uncertainty	±0.0026
Statistical Combination of Uncertainties	±0.0124
Maximum k_{eff}	0.9621
Regulatory Limiting k_{eff}	1.0000

Table 4.2.9

Summary of the Criticality Safety Analyses for Storage of Fresh Fuel Assemblies in ANO Unit 1 Region 3 Racks

Reference k_{eff}	0.9740
MCNP4a Bias	0.0009
Uncertainty in MCNP4a Bias	±0.0011
MCNP4a Statistics (95/95) Uncertainty	±0.0007
Manufacturing Tolerance Uncertainty	±0.0152
Enrichment Tolerance Uncertainty	±0.0017
Depletion Uncertainty	Not Applicable
Fuel Eccentric Positioning Uncertainty	Negative
Statistical Combination of Uncertainties	±0.0154
Maximum k_{eff}	0.9903
Regulatory Limiting k_{eff}	1.0000

Table 4.9.1

Required Soluble Boron Concentrations in the SFP Water.

Condition	Soluble Boron Required for k<1	Soluble Boron Required for k<0.95 (ppm)
Region 1: All Spent Fuel Assemblies	0	225
Region 1: Accident condition of 1 fresh fuel assembly mis-placed into a cell intended to store spent fuel	-	475
Region 1: Accident condition of 1 fresh fuel assembly mis-placed into a cell intended to remain empty	-	675
Region 2: All Spent Fuel Assemblies	0	180
Region 2: Accident condition of 1 fresh fuel assembly mis-placed into a cell intended to store spent fuel	-	450
Region 2: Checkerboard Arrangement	0	100
Region 2: Accident condition of 1 fresh fuel assembly mis-placed into a cell intended to remain empty	-	800
Region 3: All Fresh Fuel Assemblies	-	400
Region 3: Dropped fuel assembly with all fresh fuel assemblies with no poison inserts.	-	1600

Table 4.10.1

Summary of New Fuel Vault Criticality Safety Analysis

	Optimum Moderation (Figure 4.10.1)	**Flooded Moderation (Figure 4.10.1)**	**Optimum* Moderation (Figure 4.10.2)**
Initial Enrichment, wt%	4.95± 0.05	4.95± 0.05	4.20 ± 0.05
Temperature for analysis	20 °C (68 °F)	20 °C (68 °F)	20 °C (68 °F)
Reference k_{eff}	0.9639	0.9350	0.9650
Calculational bias, Δk	0.0030	0.0030	0.0030
Uncertainties			
KENO Bias	±0.0012	±0.0012	±0.0012
KENO Statistics	±0.0004	±0.0004	±0.0004
Lattice Spacing	±0.0019	±0.0016	±0.0018
Fuel Density	±0.0025	±0.0026	±0.0022
Fuel Enrichment	±0.0019	±0.0030	±0.0028
Statistical Combination	±0.0039	±0.0045	±0.0042
Total k_{eff}	0.9669 ± 0.0039	0.9380 ± 0.0045	0.9680 ± 0.0042
Maximum k-eff	0.9708	0.9425	0.9722
Regulatory Limit	0.98	0.95	0.98

* At 4.2 wt% enrichment, the flooded condition reactivity is much lower than the 4.95 wt% case.

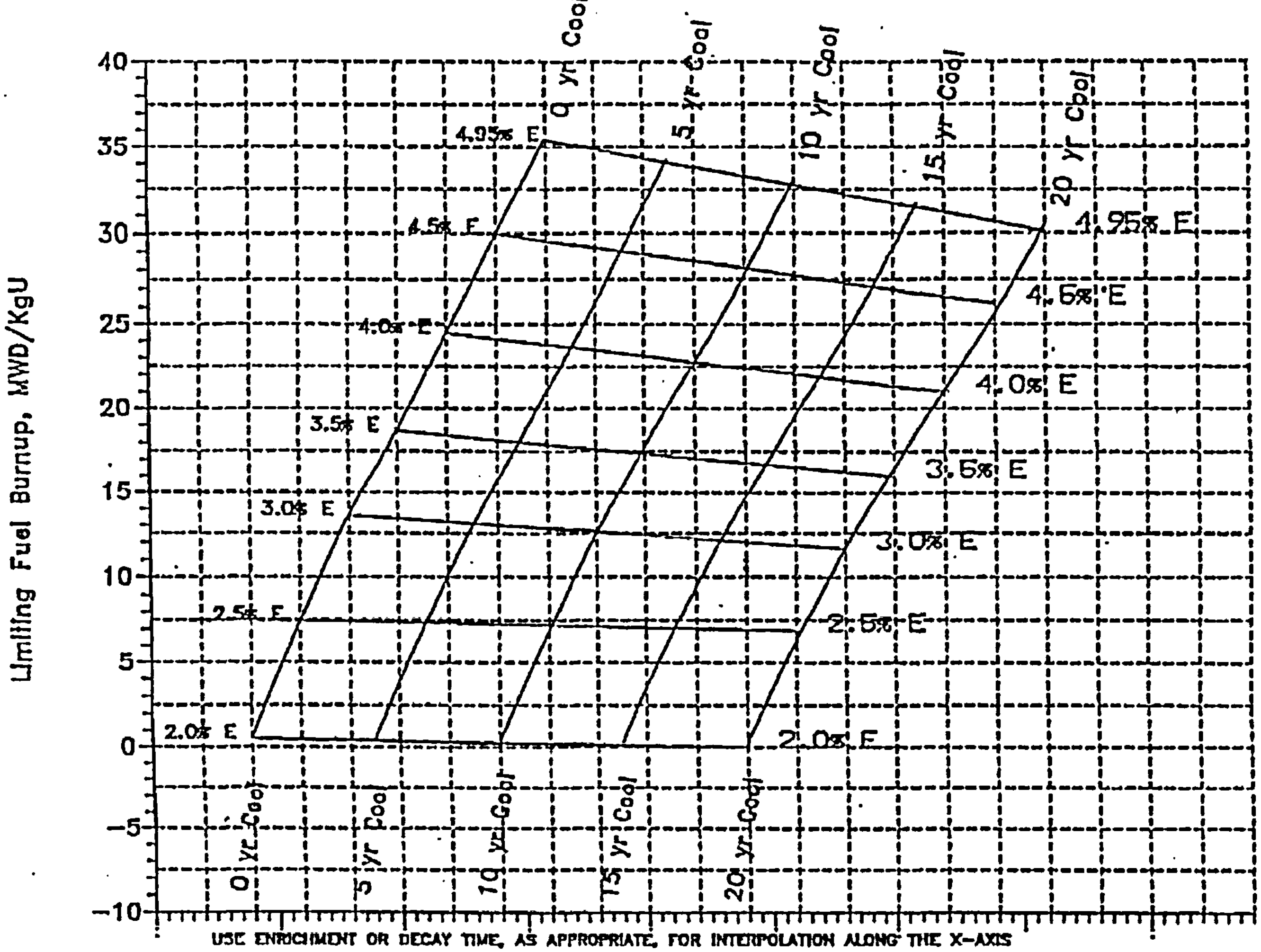

Fig. 4.1.1 3-Dimensional Plot of Minimum Fuel Burnups for Fuel in Unit 1 Region 1 for Enrichments and/or Cooling Times

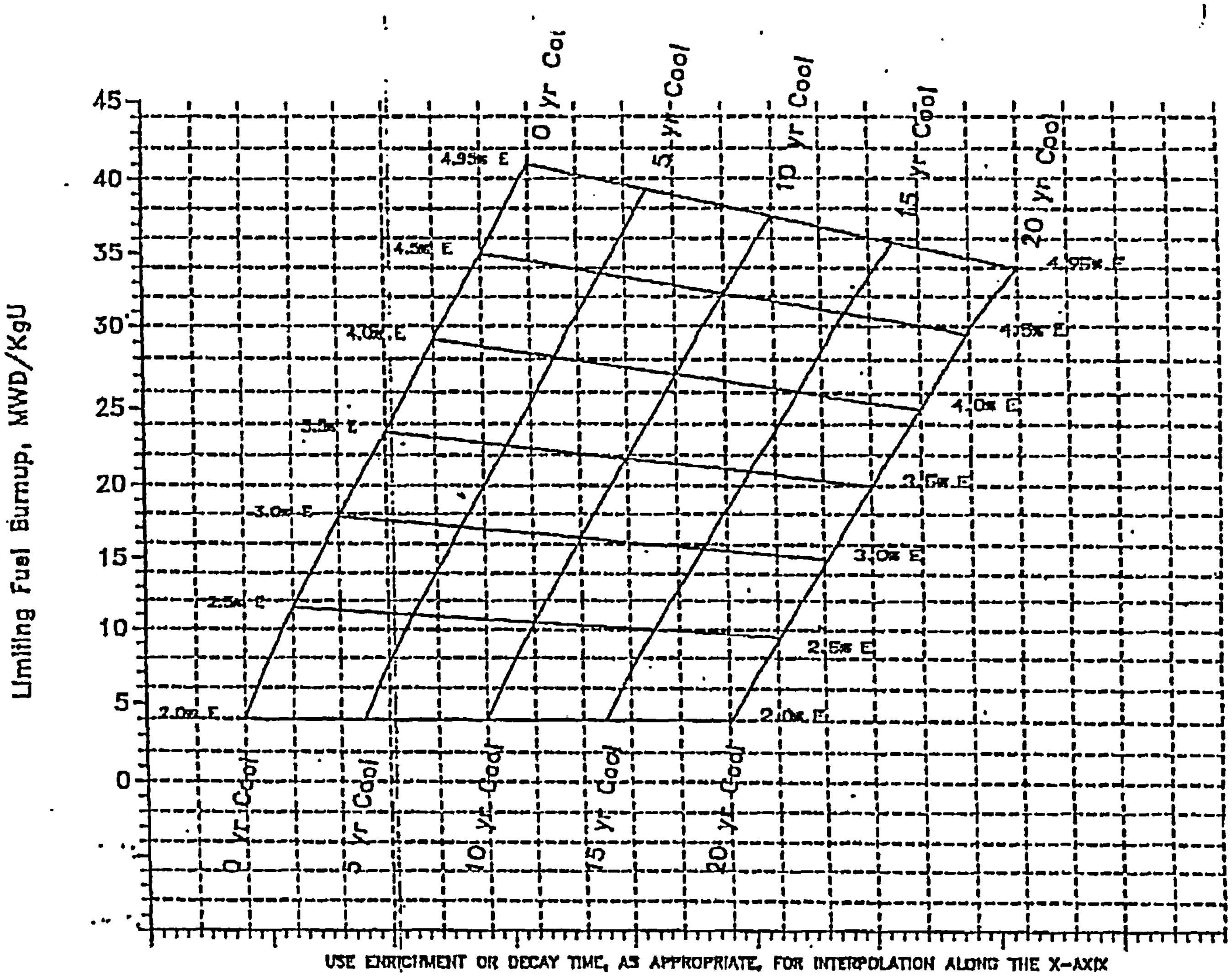

Fig. 4.1.2 3-Dimensional Plot of Minimum Fuel Burnups for Fuel in Unit 1 Region 2 for Enrichments and/or Cooling Times

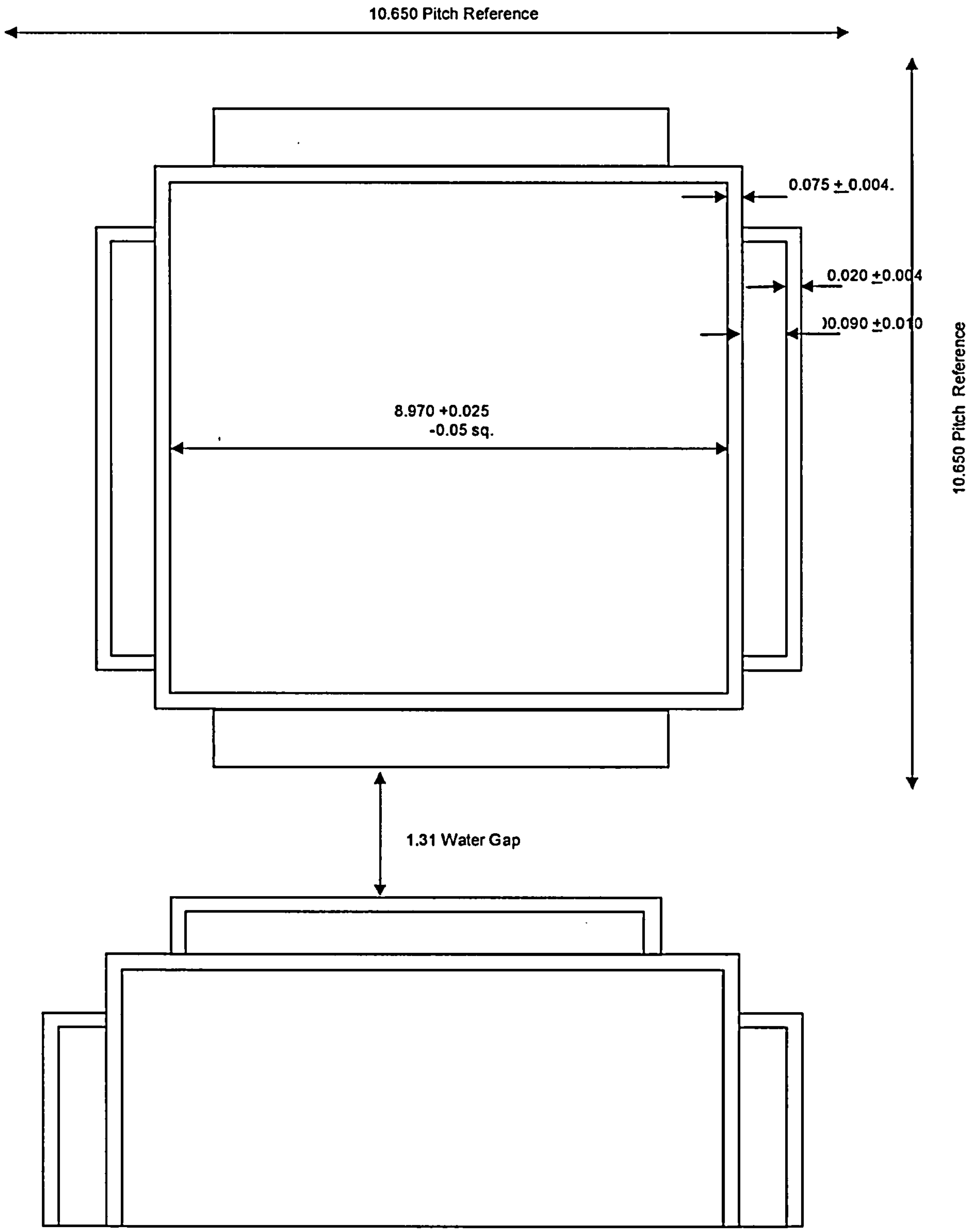

Figure 4.3.1: A Cross-Sectional View of the Calculational Model Used for the Region 1 Rack Analysis (NOT TO SCALE).

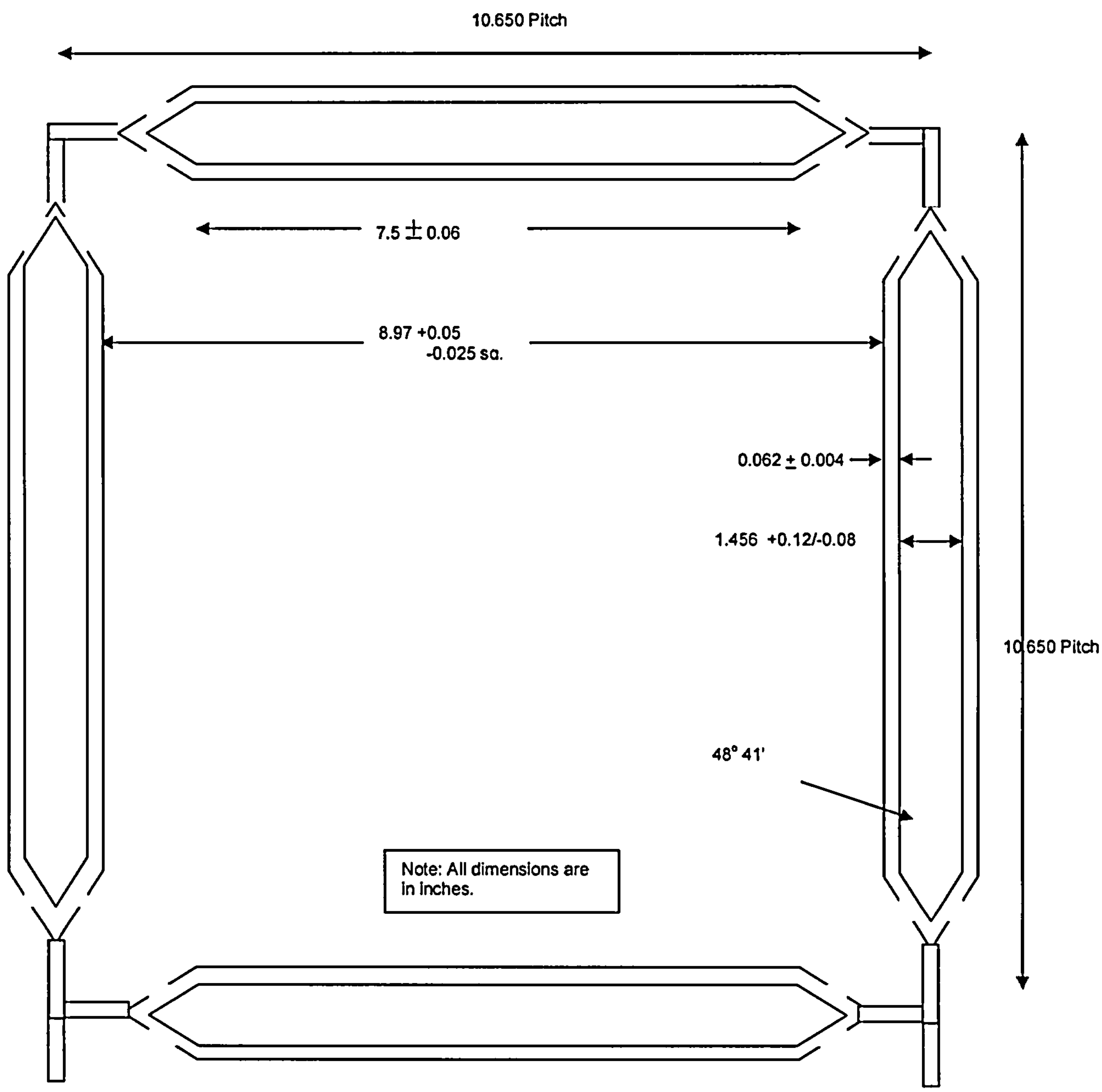

Figure 4.3.2: A Cross-Sectional View of the Calculational Model Used for the Region 2 Rack Analysis (NOT TO SCALE).

Note: The straight portion of the flux trap is modeled as 7.2 inches. In order to preserve the pitch due to a conservative reduction of the flux trap gap width from a design reference value of 1.556 inches to 1.456 inches (based on measurements which accounts for the bow in the cell walls), the cell ID was modeled as 9.07 +0.50/-0.025 inches.

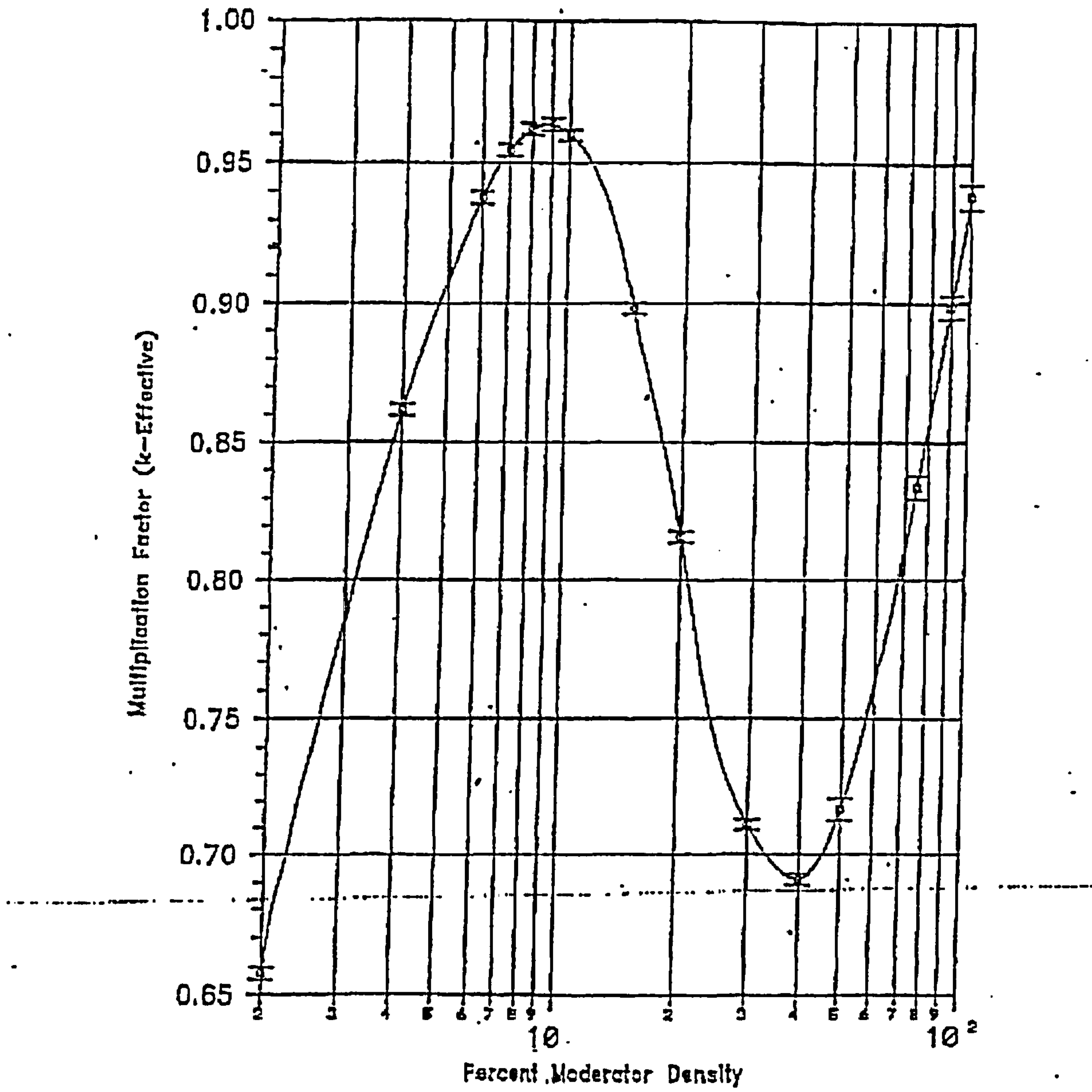

Figure 4.10.1 Reactivity of the New Fuel Vault as a Function of Moderator Density (4.95% Enrichment).

N

	A	B	C	D	E	F	G	H	K
8									
7									
6									
5			X	X	X	X	X		
4			X	X	X	X	X		
3									
2									
1									

Figure 4.10.2: Acceptable New Fuel Storage Vault Configuration for up to 4.95 wt% Enrichment Fresh Fuel.

Note: X's show the locations where fuel assemblies will not be stored.

N

	A	B	C	D	E	F	G	H	K
8									
7									
6					X				
5				X	X	X			
4				X	X	X			
3					X				
2									
1									

Figure 4.10.3: Acceptable New Fuel Storage Vault Configuration for up to 4.2 wt% Enrichment Fresh Fuel.

Note: X's show the locations where fuel assemblies will not be stored.