ATTACHMENT 1

Proposed Technical Specification Changes

North Anna Unit 1

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 280°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1780 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.3 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

ATTACHMENT 2

Proposed Technical Specification Changes

North Anna Unit 2

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1780 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.3 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9957 ± 10 cubic feet at a nominal T_{avg} of 525°F.

ATTACHMENT 3

Safety Evaluation for

Increased Fuel Enrichment

1.0 INTRODUCTION

The current North Anna Technical Specifications (TS) limit of 4.1 w/o U^{235} must be raised to allow higher fuel enrichments than that currently used (4.0 w/o U^{235}) in North Anna reload core designs. An increase in the enrichment TS limit to 4.3 w/o U^{235} will allow accomplishment of fuel management plans to increase batch average discharge burnups to the current licensed limit of 45,000 MWD/MTU while maintaining cycle energy requirements. An enrichment of 4.3 w/o U^{235} was chosen for the new fuel storage criticality analysis because the current spent fuel storage racks are currently licensed to an enrichment of 4.3 w/o U^{235} (Reference 1).

The safety impact of operation of the North Anna and Surry units with high burnup fuel was previously addressed by Virginia Electric and Power Company in References 2 through 5. The NRC subsequently approved, Reference 6, operation of the Surry and North Anna fuel to a batch average discharge burnup of 45,000 MWD/MTU. Westinghouse generically addressed the impact of extended burnup on the design and operation of Westinghouse fuel (Reference 7). In addition, the NRC had an independent assessment conducted, Reference 8, of the environmental and economic impacts of the use of extended burnup fuel in light water power reactors. The overall findings of this assessment were that no significant adverse effects would be generated by increasing the present batch-average burnup level to values of 50,000 MWD/MTU or above, as long as the maximum rod average burnup of any rod is no greater than 60,000 MWD/MTU. Since the conclusions of these evaluations concerning the impact of extended burnup fuel are valid for an enrichment of 4.3 w/o U^{235} and since the spent fuel storage facility is currently licensed to 4.3 w/o, this submittal only addresses the impact of increased enrichment on the requirements for new fuel storage racks.

The specific 10 CFR 50 Appendix A General Design Criteria for new fuel storage facilities are listed in Section 9.1.1 of the Standard Review Plan (NUREG-0800). Since no physical modifications are being made to the current new fuel racks, this analysis only addresses the impact of the increased enrichment on the requirement of subcriticality under normal and postulated abnormal rack conditions (General Design Criterion 62). The highest K-effective allowable by Section 9.1.1 of NUREG-0800 for all conditions is 0.98.

The computer modeling of the storage racks was performed in 3-D to minimize unnecessary conservatism and uncertainty. All K-effective calculations were performed with the Monte-Carlo program KENO V.a (Reference 9) within the SCALE (Reference 10) package. The SCALE package automatically processes cross sections through NITAWL and BONAMI to create a set of resonance self shielded cross sections for use by KENO. Because all calculations for this analysis were made using a discrete pin representation, no spatial self shielding was performed prior to the KENO execution. The cross section set chosen was the 27 group ENDF/B-IV data contained in the SCALE package. Sufficient neutron histories were run for each case to limit the statistical uncertainties in the K-effective to less than 0.4% ΔK/K.

2.0 MODEL DATA

The new fuel storage area at North Anna consists of nine parallel rows of storage racks with a total capacity of 126 fuel assemblies (Figure 2.1). Each storage location consists of a square 9 inch (inside measure) stainless steel box 165 inches tall with walls 1/8 inch thick. The storage area walls and floor are concrete. A steel grating at the top prevents accidental placement of an assembly between storage cans. The storage area is normally dry.

Several fuel assembly and rack components have been neglected in this model for simplicity and conservatism. Assembly top and bottom nozzles (SS-304), grids (Inconel), sleeves (SS-304), and all storage rack structural materials other than the storage can itself are modeled as void or moderator regions. These omissions are all conservative from a criticality standpoint because steel and Inconel are both strong neutron absorbers.

Fuel assembly dimensions and material data are provided in Table 2.1. A top view of the storage area is given in Figure 2.1. Figure 2.2 shows the side view as modeled. Undesignated areas are air space under normal conditions.

TABLE 2.1

NORTH ANNA 17 x 17 FUEL ASSEMBLY DATA

Fuel Enrichment	--	4.3 w/o U^{235}
Assembly Pitch	--	8.466 in.
Pellet Diameter	--	0.3225 in.
Diametral Gap	--	0.0065 in.
Clad Thickness	--	0.0225 in.
Clad O.D.	--	0.3740 in.
Pellet Material	--	95% th. dens. UO_2
Clad Material	--	Zircaloy-4
Fuel Rod Pitch	--	0.4960 in.
Active Fuel Length	--	144.0 in.
Fuel Rods/Assembly	--	264
Guide Tubes/Assembly	--	25
Guide Tube Material	--	Zircaloy-4
Guide Tube O.D.	--	0.482 in.
Guide Tube I.D.	--	0.450 in.

FIGURE 2.1
2-D STORAGE AREA DIAGRAM

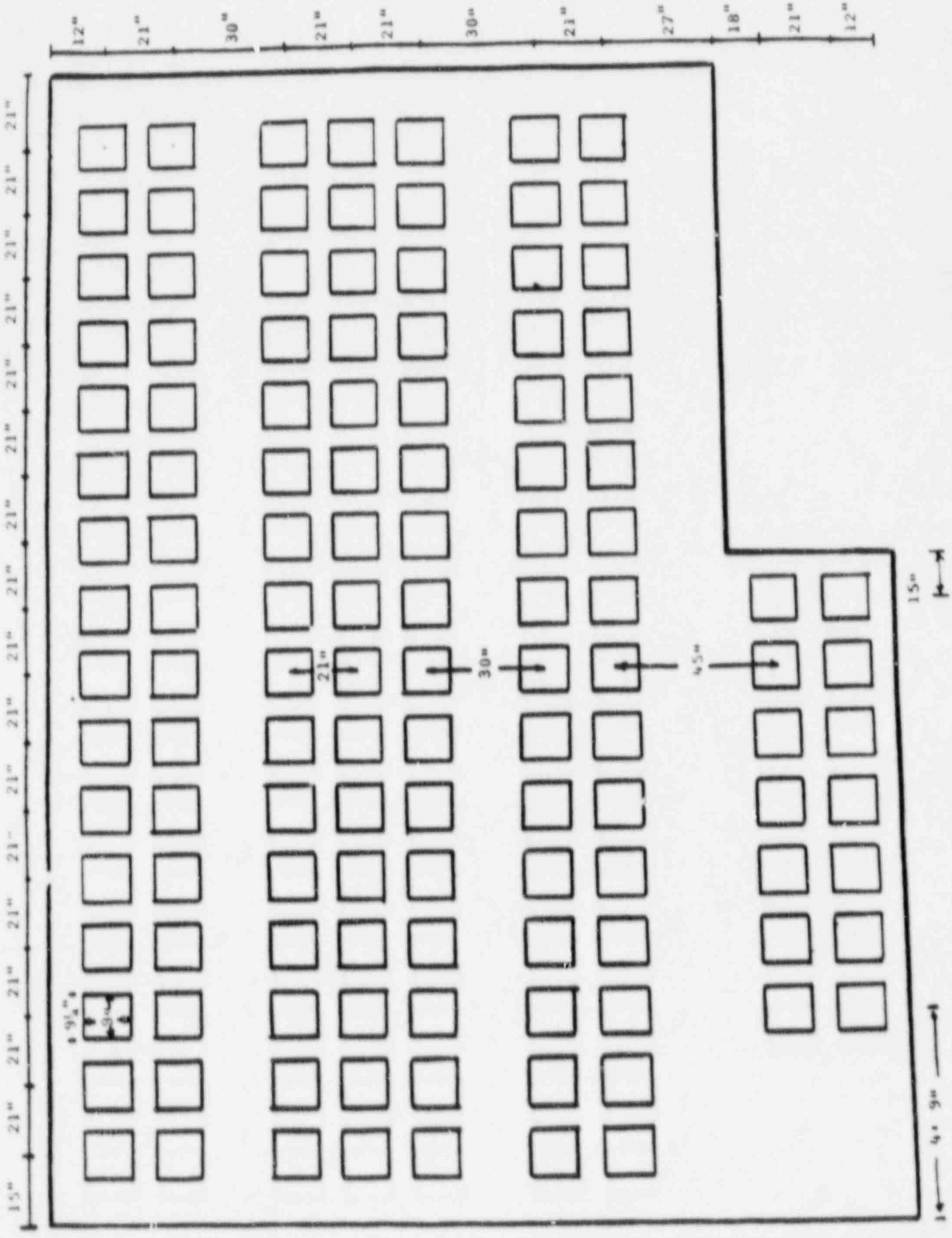
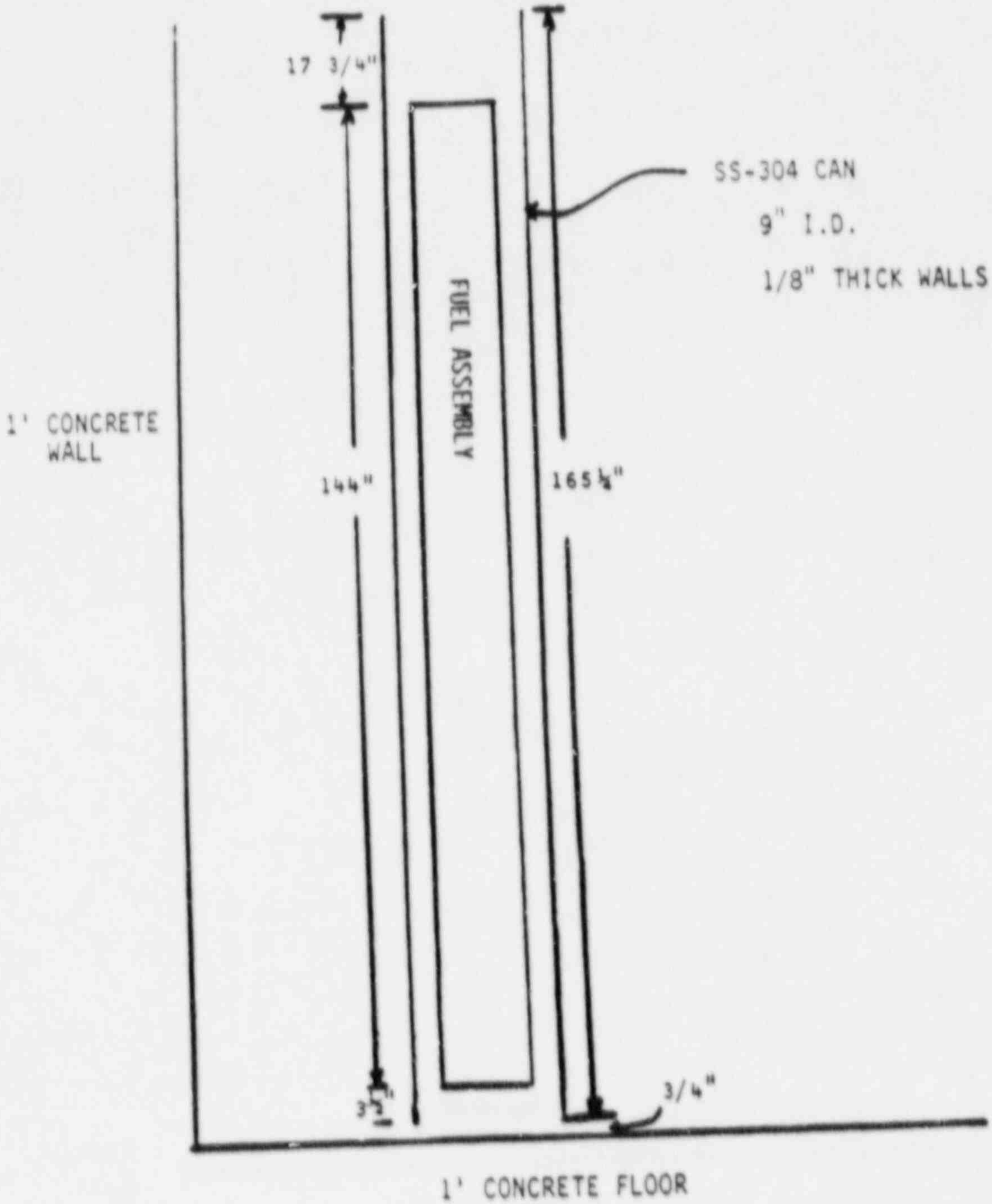


FIGURE 2.2
AXIAL STORAGE AREA DIAGRAM



3.0 CONDITIONS MODELED

3.1 Normal Configuration

The base condition for the analysis consisted of a fully loaded storage area of 126 fresh 4.3 w/o U^{235} enriched assemblies centered nominally in the storage cans. The air regions in the storage area were modeled as water vapor at 10^{-8} g/cc density.

3.2 Moderator Density Variation (Optimum Moderation)

Normal air humidity variations from dry conditions to heavy fog can result in water densities ranging from 0 to .0025 g/cc (Reference 11). In addition, fire or a pipe break can result in flooding of the storage area by foam or water of many possible densities. To allow for these conditions, the air regions in the storage area were assigned water densities ranging from 10^{-6} g/cc to 0.998 g/cc.

3.3 Fuel Pitch Variation

Eccentric assembly positioning or a seismic event can lead to small assembly pitch changes. Assuming the rack does not deform leads to a maximum pitch change for any two assemblies of ±0.57 inches. Although any pitch changes are likely to be random, the effect of pitch reduction on K-effective has been conservatively determined by reducing the pitch of all the storage locations by 0.5 and 1.0 inch. For the pitch variation, the KENO model was simplified from the base version used for all other calculations. Rather than modeling the non-storage rectangular area next to the short row of assemblies explicitly, the area was represented as a block of concrete. The moderator and concrete walls surrounding the fuel were modeled as surrounding cuboids around a single combined fuel array. This simplified model made pitch changes much easier and produced eigenvalues within 2 standard deviations of the more complicated representation. Results obtained with this model were used only to determine the change in K-effective for a change in fuel pitch.

3.4 Fuel Drop Accident

A dropped assembly could result in the fuel being compacted within the storage cell. To conservatively model this accident, the fuel pellet diameter of all assemblies in the rack was increased 10%. Calculations were performed assuming no change in assembly height and with a change in assembly height which preserves the total fuel volume (both at 95% theoretical density UO_2). The compaction effect was determined at two moderator densities. In the compaction model the fuel was assumed to contact and radially expand the clad (ie., clad thickness was preserved).

4.0 RESULTS

The K-effective for each rack condition analyzed is listed in Table 4.1. Results are summarized for each configuration below.

4.1 Worst Case Normal Configuration

The base K-effective for the nominally loaded 4.3 w/o U^{235} dry storage area is 0.443 using the 27 group ENDF/B-IV cross section set. Because normal humidity changes can result in moderator densities up to 0.0025 g/cc and fuel can be eccentrically placed in the rack, the difference in K-effective caused by these changes must be added to the base value. For eccentric assembly placement the conservative assumption of a uniform reduction in the pitch of the entire rack of 0.57 inch (interpolated from 0.5 and 1.0 inch results at 0.07 moderator density) results in an increase in K-effective of 0.009 $\Delta K/K$. A moderator density increase from 10^{-8} to 0.01 g/cc increases K-effective by 0.12 $\Delta K/K$. The worst cask normal K-effective (excluding calculational uncertainty and bias) is therefore 0.572.

4.2 Worst Case Abnormal Configuration

The worst case abnormal configuration is considered to be equal to the worst case normal K-effective plus the maximum difference caused by a single accident condition.

K-effective as a function of moderator density reaches a peak at 0.07 g/cc water density (optimum moderation case) as shown in Figure 4.1. The K-effective increase from 0.01 g/cc to 0.07 g/cc is + 0.312 $\Delta K/K$. Note that at a moderator density of 1.0 g/cc the K-effective is nearly identical to the peak value at 0.07 g/cc. The second peak is due to increased neutron reflection at the storage area boundaries which overcomes any negative impact on K-effective due to overmoderation at moderator densities above 0.07 g/cc. This effect might not be observed in simplified 2D or 1D models.

Fuel compaction due to dropping fuel assemblies results in a gain of only 0.01 $\Delta K/K$ (0.027 $\Delta K/K$ if the fuel height change is neglected). No accounting for pitch changes are necessary because the change in K-effective due to the maximum possible pitch reduction (assuming the racks are not deformed in a seismic event) has already been accounted for in the normal configuration K-effective. The worst case abnormal configuration (exclusive of uncertainty or bias) is 0.884.

4.3 Effect Of Calculational Uncertainty And Bias

The statistical uncertainty of the KENO new fuel storage rack calculations is less than ± .004 at the 95% confidence level. Calculations to benchmark KENO-V.a using the 27 group ENDF/B-IV cross sections against critical experiments indicated a consistent bias of -0.011 $\Delta K/K$ in the predicted versus experimental criticality. This bias is consistent with other reported values based on a larger number of critical experiments (Reference 12), although the minimum calculated K-effective in Reference 12 is lower by more than 1% (minimum K-eff = 0.974). Use of the HANSEN-ROACH cross sections resulted in much greater scatter and uncertainty in the benchmark eigenvalues and led to the decision to use the ENDF/B-IV set. Adding the average bias and uncertainty to the worst case values from above yields the following net results:

Assuming the average bias:

Worst case normal configuration K-eff:	0.587
	(0.572 + 0.004 + 0.011)
Worst case abnormal configuration K-eff:	0.899
	(0.884 + 0.004 + 0.011)
K-eff limit:	0.980
Margin:	0.081 $\Delta K/K$
	(0.980 - 0.899)

Assuming the worst case bias from Reference 12:

Worst case normal configuration K-eff:	0.602
	(0.572 + 0.004 + 0.026)
Worst case abnormal configuration K-eff:	0.914
	(0.884 + 0.004 + 0.026)
K-eff limit:	0.980
Margin:	0.066 $\Delta K/K$
	(0.980 - 0.914)

Note that enough margin exists that the criticality criteria can be satisfied regardless of whether the average bias or worst case bias from Reference 12 is applied.

TABLE 4.1

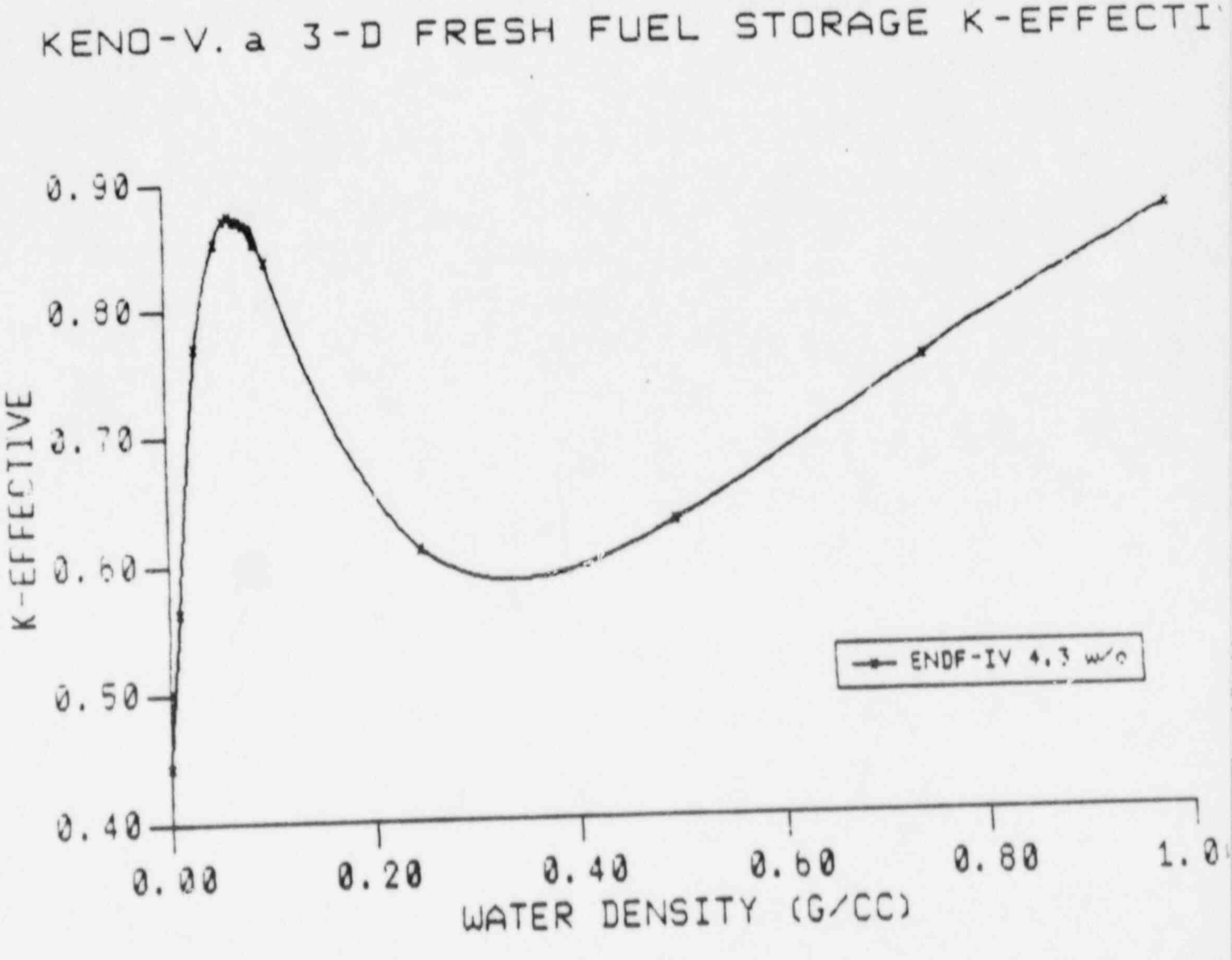
NORTH ANNA NEW FUEL STORAGE RACK K-EFFECTIVE

4.3 W/O U^{235}

CASE TYPE	MOD.	K-EFF		S.E.	COMMENTS
BASE	10-8	0.44320	±	0.00108	Nominal base case
ALT. ENRICH	10-8	0.43735	±	0.00106	4.1 w/o sensitivity case
DENSITY	10-6	0.44264	±	0.00112	
DENSITY	10-4	0.44267	±	0.00111	
DENSITY	0.01	0.56330	±	0.00123	
DENSITY	0.03	0.76922	±	0.00132	
DENSITY	0.05	0.85394	±	0.00143	
DENSITY	0.06	0.87233	±	0.00151	
DENSITY	0.065	0.87425	±	0.00159	
DENSITY	0.07	0.87535	±	0.00150	
DENSITY	0.075	0.87006	±	0.00152	
DENSITY	0.08	0.86746	±	0.00132	
DENSITY	0.09	0.85361	±	0.00135	
DENSITY	0.10	0.83763	±	0.00130	
DENSITY	0.25	0.61097	±	0.00133	
DENSITY	0.50	0.62982	±	0.00145	
DENSITY	0.75	0.75535	±	0.00165	
DENSITY	1.00	0.87167	±	0.001[illegible]	
PITCH BASE	10-8	0.44265	±	0.00114	Simplified for pitch change
PITCH -0.5 IN.	10-8	0.44255	±	0.00103	
PITCH -1.0 IN.	10-8	0.44155	±	0.00110	
PITCH BASE	0.07	0.87200	±	0.00151	Simplified for pitch change
PITCH -0.5 IN.	0.07	0.88048	±	0.00139	
PITCH -1.0 IN.	0.07	0.88139	±	0.00153	
DROP FUEL	0.065	0.89166	±	0.00144	144 inch fuel height
DROP FUEL	0.065	0.86610	±	0.00148	119 inch fuel height
DROP FUEL	10-8	0.47018	±	0.00102	144 inch fuel height
DROP FUEL	10-8	0.45327	±	0.00112	119 inch fuel height

NOTES: Mod. density is KENO vol. frac. Nominal density is 0.9982 g/cc.

FIGURE 4.1
K-EFFECTIVE VERSUS MODERATOR DENSITY



5.0 CONCLUSIONS

The results discussed in Section 4 clearly indicate, that for a fuel enrichment of 4.3 w/o U^{235}, the North Anna new fuel storage area meets the criticality limit of K-effective <0.98 and is safe under the criticality specifications set forth in the Standard Review Plan (NUREG-0800).

6.0 PROPOSED TECHNICAL SPECIFICATION CHANGES

In Section 5.3.1 of both the North Anna Unit 1 and Unit 2 Technical Specifications, replace the last sentence with the following wording:

> Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.3 weight percent U-235.

10 CFR 50.59 EVALUATION

The proposed changes do not result in an unreviewed safety question as defined in 10 CFR 50.59. The results of this evaluation can be stated as follows:

1. No increase in the probability of occurrence or consequences of an accident will occur as a result of the proposed increase in enrichment. The only accident scenarios for which the probability of occurrence are potentially affected by fuel enrichments involve criticality events during fuel handling and storage. The enclosed criticality safety analysis demonstrates that K-effective during handling and storage of new fuel is low enough to ensure subcriticality during postulated accident conditions. In addition, the criticality safety analysis approved by Reference 1 demonstrated that the calculated K-effective during handling and storage of spent fuel was low enough to ensure subcriticality during postulated accident conditions. The probability of occurrence of criticality during fuel handling or storage is therefore not increased. Since subcriticality is maintained, no releases would result from the above handling and storage accident scenarios. In addition, since the burnup limit will not be increased beyond that approved in Reference 6, radiological consequences of the accidents discussed in References 6, 7 and 8 will not be increased.

2. The possibility of a new or different kind of accident from any previously evaluated is not created. The only potential impact of increased enrichment upon fuel storage and handling involves the potential for criticality which has been addressed above.

3. The margin of safety is not reduced. The enclosed criticality analysis demonstrates that the limit on K-effective (0.98) as defined in Section 9.1.1 of NUREG-0800 is met. Therefore, there is adequate margin to ensure subcriticality during the storage and handling of new fuel and the requirements of 10 CFR 50 Appendix A General Design Criterion 62 are satisfied. The safety analysis approved in Reference 1 provides the same assurance for spent fuel.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed change does not involve a significant hazards consideration because operation of North Anna 1 and 2 in accordance with the change would not:

1. Involve a significant increase in the probability or consequences of accidents previously evaluated. The only accident scenarios for which the probability of occurrence are potentially affected by fuel enrichments involve criticality events during fuel handling and storage. The criticality safety analyses demonstrates that K-effective during fuel handling and storage of new fuel is low enough to ensure subcriticality during postulated accident conditions. In addition, the criticality safety analysis approved by Reference 1 demonstrated that the calculated K-effective during fuel handling and storage of spent fuel was low enough to ensure subcriticality for all postulated accident conditions. The probability of occurrence of criticality during fuel handling or storage is therefore not increased. Since subcriticality is maintained, no releases would result from the fuel handling and storage accident scenarios. In addition, since the burnup limit will not be increased beyond that approved in Reference 6, the radiological consequences of the accidents discussed in References 6, 7 and 8 will not be increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated. The only potential impact of increased enrichment upon fuel storage and handling involves the potential for criticality which has been addressed above.

3. The proposed amendment does not involve a significant reduction in the margin of safety. The criticality analysis demonstrates that there is adequate margin to ensure subcriticality of the fuel during storage and handling of new fuel. The safety analysis approved in Reference 1 provides the same assurance for spent fuel.

Therefore, pursuant to 10 CFR 50.92, based on the above consideration, it has been determined that these changes do not constitute a significant safety hazards consideration.

REFERENCES

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4. Letter from B. R. Sylvia (Virginia Electric and Power Company) to H. R. Denton (NRC), Serial No. 195, March 26, 1981.

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8. "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Pacific Northwest Laboratory, NUREG/CR-3009, February, 1988.

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11. E. T. Tomlinson and C. L. Brown, "Nuclear Criticality Safety Considerations in Design of Dry Fuel Assembly Storage Arrays," Nuclear Technology, Vol. 63, November 1983.

12. T. L. Sanders, R. M. Westfall, and R. H. Jones, "Feasibility and Incentives for the Consideration of Spent Fuel Operating Histories in the Criticality Analysis of Spent Fuel Shipping Casks," SAND87-0151, TTC-0713, August, 1987.