VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

November 5, 1997

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 97-614
NL&OS/GDM R0
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS NO. 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE
FOR INCREASED ENRICHMENT OF RELOAD FUEL

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company requests amendments, in the form of changes to the Technical Specifications for Operating License Nos. DPR-32 and DPR-37, Surry Power Station Units 1 and 2, respectively. The proposed change will revise the Technical Specifications to increase the maximum allowable fuel enrichment from 4.1 to 4.3 weight percent U^{235}. This will permit discharge burnups that are more compatible with the current lead rod burnup limit of 60,000 MWD/MTU, and will result in fuel cycle cost savings while continuing to satisfy cycle energy requirements. A discussion of the proposed change is included as Attachment 1.

The proposed change has been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee. It has been determined that the proposed change does not involve an unreviewed safety question as defined in 10 CFR 50.59 or create a significant hazards consideration as defined in 10 CFR 50.92. The proposed Technical Specifications change and the basis for the no significant hazards determination are included as Attachments 2 and 3, respectively.

The first implementation of the higher fuel enrichment is scheduled to be in Surry 1 Cycle 16, which will begin operation in November 1998. To support the fuel delivery schedule for this cycle, we request NRC review and approval of the requested change by July 31, 1998. If the review will be delayed for any reason, it will be necessary to

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modify our fuel and burnable absorber orders with our vendors, to ensure that the Surry 1 Cycle 16 core design complies with the existing Technical Specifications enrichment limits.

Should you have any questions or require additional information, please contact us.

Very truly yours,

James P. O'Hanlon
Senior Vice President - Nuclear

Attachments
1. Discussion of Change
2. Proposed Technical Specifications Change
3. Significant Hazards Consideration Determination

cc: U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

Commitment Summary

1. The commitments made in this letter are as indicated in the proposed Technical Specifications change.

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. P. O'Hanlon, who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 5th day of November, 1997.

My Commission Expires: March 31, 2000.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Discussion of Change

DISCUSSION OF CHANGES

INTRODUCTION

The fuel burnup at Surry Units 1 and 2 is currently limited to batch average burnups of 50,000 MWD/MTU or above, provided that the maximum rod average burnup of any fuel rod is no greater than 60,000 MWD/MTU. The actual burnups achieved depend on the number of assemblies loaded each cycle, but with the current fuel enrichments the Surry batch average burnups are typically about 45,000 MWD/MTU, with lead rod burnups ranging from about 49,000 to 56,000 MWD/MTU.

Virginia Electric and Power Company proposes to increase the maximum fuel enrichment for Surry Units 1 and 2 from the current Technical Specifications limit of 4.1 weight percent U^{235}, to 4.3 weight percent U^{235}. This will permit fuel discharge burnups more compatible with the current lead rod burnup limit of 60,000 MWD/MTU, resulting in fuel cycle cost savings while continuing to satisfy our cycle energy requirements. A similar change to the North Anna Unit 1 and North Anna Unit 2 Technical Specifications was approved by the NRC in 1990 (Reference 1).

BACKGROUND

In May, 1980, the Virginia Electric and Power Company requested an increase in the maximum fuel enrichment for Surry Units 1 and 2 from 3.6 to 4.1 weight percent U^{235}. This request was made to support an eventual increase in the discharged fuel burnup. Following an interim change of the enrichment limit to 3.7 weight percent U^{235} during the review process, a maximum fuel enrichment of 4.1 weight percent U^{235} was approved by the NRC in January, 1982, for batch average burnups up to 37,000 MWD/MTU (Reference 2). This burnup limit was imposed while further review of the impact of operation to high burnups was conducted by the NRC.

When the batch average burnup limit for both the Surry and North Anna Power Stations was subsequently increased to 45,000 MWD/MTU in April, 1984 (Reference 3), NRC review of fuel vendor topical reports on extended fuel burnup was not yet complete. Subsequent to approval of our fuel vendor's high burnup topical report (Reference 4) and publication of NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors" (Reference 5), the Virginia Electric and Power Company requested an increase in the Surry and North Anna fuel burnup limits to the levels approved in the Westinghouse topical report. In December, 1993, the NRC approved use of batch average burnups of 50,000 MWD/MTU, or above, provided that the maximum rod average burnup of any fuel rod is no greater than 60,000 MWD/MTU (References 6 and 7). During the changes in burnup limits since 1982, the fuel enrichment limit at the Surry Power Station has remained at 4.1 weight percent U^{235}.

TECHNICAL SPECIFICATIONS CHANGES

Two of the Surry Units 1 and 2 Technical Specifications are affected by the proposed change:

1) Technical Specification 5.3.A.3

   The Specifications on the Reactor Core will be changed to increase the maximum enrichment of the reload fuel from 4.1 to 4.3 weight percent U^{235}.

2) Technical Specification 5.4.B

   The specification for the spent fuel storage rack, which is common to Surry Units 1 and 2, will similarly be modified to increase the maximum enrichment of the fuel which may be stored in the spent fuel racks from 4.1 to 4.3 weight percent U^{235}. A typographical error in this specification is also being corrected, to note that K-effective is required to be less than or equal to 0.95 in the spent fuel pool ("≤" symbol is being added). This Technical Specification also includes a figure (Figure 5.4-1) which defines the fuel assemblies which are acceptable for storage in Region 1 of the Surry spent fuel pool (the locations adjacent to the Fuel Building Trolley Load Block) during spent fuel cask handling. This figure is also being modified to include the potential use of fuel with the higher initial enrichment. Other than modification of Figure 5.4-1, no changes are required to the administrative controls which govern the placement of fuel into Region 1 of the spent fuel pool.

SAFETY SIGNIFICANCE SUMMARY

The use of a higher fuel enrichment for Surry Units 1 and 2 has the potential to impact criticality safety analyses for fuel handling and storage. Calculations were performed to address the impact on the new and spent fuel storage areas, which are common to Surry Units 1 and 2. It was demonstrated that the Surry new and spent fuel storage areas meet the criticality specifications set forth in Sections 9.1.1 and 9.1.2, respectively, of the Standard Review Plan (NUREG-0800). Use of a slightly higher initial fuel enrichment in Surry fuel assemblies will not affect the inputs to any Surry safety analysis, nor will the consequences of accident scenarios described in the Surry UFSAR be affected. Other areas which might be affected by a change in the fuel enrichment were reviewed, and no adverse impacts were identified. Increasing the maximum initial fuel enrichment for Surry Units 1 and 2 will not result in an unreviewed safety question as defined in 10 CFR 50.59, and will not constitute a significant hazards consideration as defined in 10 CFR 50.92.

TECHNICAL AND SAFETY EVALUATION

Virginia Electric and Power Company has evaluated the potential impacts of operating with higher fuel enrichments at Surry Units 1 and 2. As described in further detail below, previous evaluations of the safety impact of operation with high burnup fuel were based on assessments that were valid for enrichments which exceed the requested maximum enrichment for Surry. Therefore the proposed enrichment increase will not affect the types or amounts of radiological effluents that may be released offsite. The use of a higher fuel enrichment also has the potential to impact criticality safety analyses for fuel handling and storage. Calculations were performed to demonstrate that the General Design Criteria for fuel storage facilities which are listed in Sections 9.1.1 and 9.1.2 of the Standard Review Plan (Reference 8) will be satisfied for the higher fuel enrichment under normal and postulated abnormal rack conditions. These calculations are described in more detail below.

1. Environmental Effects of Higher Enrichment Fuel

The safety impact of operation of the Surry units with high burnup fuel was previously addressed in References 9 through 18. Westinghouse has generically addressed the impact of extended burnup on the design and operation of Westinghouse fuel (Reference 4). In addition, the NRC had an independent assessment conducted (Reference 5) 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 batch average burnup level to values of 50,000 MWD/MTU or above, as long as the maximum rod average burnup of any fuel rod is no greater than 60,000 MWD/MTU. The conclusions of these evaluations concerning the impact of extended burnup fuel are valid for an enrichment of 4.3 weight percent U^{235}. In 1993, the NRC approved operation of current enrichment Surry fuel to burnup levels consistent with these findings (References 6 and 7).

The source terms for gap and core inventories which are used to determine releases from fuel during various accident scenarios are a function of the number of fissions which have occurred (i.e., the power level and burnup) rather than the amount of fissionable material initially present in the fuel. The primary benefit of having a higher initial fuel enrichment is that the same power level can be produced for a longer time before the unit must be refueled. However, the operating power and fuel burnup limits will not be changed for Surry when the maximum fuel enrichment is increased. It is therefore concluded that increasing the maximum fuel enrichment for Surry Units 1 and 2 will not adversely affect the types or amounts of any radiological effluents that may be released offsite.

In addition, the Virginia Electric and Power Company has reviewed its current and proposed fuel use at the Surry Power Station, and has determined that the "NRC Assessment of the Environmental Effects of Transportation Resulting from Extended Fuel Enrichment and Irradiation" as provided in the Federal Register (53 FR 30355) is properly applicable to Surry. At the current Technical Specifications fuel enrichment limit of 4.1 weight percent U^{235}, batch average burnups of about 45,000 MWD/MTU are achievable at Surry. As a result of the proposed increase in the enrichment limit to 4.3 weight percent U^{235}, the batch average burnups

may increase slightly, approaching about 50,000 MWD/MTU, but not exceeding a rod average burnup of 60,000 MWD/MTU for any fuel rod. Therefore, both the current enrichment and burnup levels and the proposed enrichment and burnup levels are within the limits for these parameters assumed in the NRC staff analysis performed pursuant to 10 CFR 51.52(b) and documented in the Federal Register (53 FR 30355).

It may be noted that the current generation Surry fuel to which the proposed enrichment increase would apply also uses ZIRLO for the fuel rod cladding, rather than using Zircaloy clad fuel as discussed in 10 CFR 51.52(a)(2). As the properties, including strengths, of these two zirconium-based cladding materials are very similar, there is no increase in the likelihood of a radiological release in a transportation accident involving fuel with ZIRLO cladding. Reference 19 (WCAP-12610) also indicates that comparable core inventories are predicted by Westinghouse for operation of fuel with the two cladding materials to a given burnup level. There is therefore no change in the radiological impact of transportation accidents with ZIRLO, versus Zircaloy, clad fuel. In addition, the ZIRLO clad fuel at Surry replaces Zircaloy clad fuel on a one-for-one basis, so the number of shipments of fresh fuel, spent fuel, or low level waste to or from the facility are unaffected. Consequently, there is no change to either the likelihood of an accident or the non-radiological consequences of transportation accidents resulting from the use of ZIRLO in current Surry fuel. Virginia Electric and Power company concludes that the use of fuel with ZIRLO cladding does not affect the applicability of Table S-4 in 10 CFR 51.52(c) to transportation of Surry fuel.

Virginia Electric and Power Company has also reviewed the NRC Staff's Environmental Assessment contained in the enclosure to an NRC letter dated April 21, 1986, which transmitted Amendments No. 76 and 65 of the North Anna Unit 1 and Unit 2 operating licenses, respectively. This assessment related to the transshipment of spent fuel from Surry to North Anna. The report stated that the environmental impact of the proposed transshipment of spent fuel from Surry to North Anna was well within the scope of Table S-4 as set forth in 10 CFR 51.52(c), and need not be addressed on a site specific basis. The Environmental Assessment concluded that the radiological impact on the environment of the proposed transshipment would be less than that shown in Table S-4 by a factor of at least 30, and is accordingly well within the scope of Table S-4. This evaluation is therefore not impacted by either the proposed increase in enrichment for Surry fuel, or by the use of ZIRLO cladding in current Surry fuel.

Since the enrichment limits and discharge burnups for Surry fuel are below the limits assumed for these parameters in the Federal Register (53 FR 30355) analysis, and the use of current fuel with ZIRLO cladding does not affect the probability or consequences of accidents considered by the analysis, Virginia Electric and Power Company concludes that this analysis is applicable to the Surry facilities and our intended fuel use. Therefore, Virginia Electric and Power Company adopts the use of the assessment to the environmental effect of the transportation of high burnup fuel, provided in the Federal Register (53 FR 30355), to satisfy the requirements of 10 CFR 51.52(b).

2. Impact on New Fuel Storage Area

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; Reference 8). Since no physical modifications are being made to the current Surry new fuel racks, this analysis addresses only 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 allowed by Section 9.1.1 of NUREG-0800 for the new fuel racks under all conditions is 0.98.

A single analysis was performed for the Surry and North Anna new fuel storage racks, since the new fuel storage areas for the two power stations are of the same design, both geometrically and materially. This analysis, which was previously submitted for North Anna in Reference 20, is summarized below. The calculations performed incorporated nominal fuel assembly dimensions for North Anna's 17x17 fuel design. By comparison, Surry Units 1 and 2 use fuel with a 15x15 array of slightly larger diameter fuel rods. Although the two fuel assembly designs have comparable overall dimensions, for the same nominal fuel density the Surry fuel design contains approximately 1% less UO_2 per assembly. Calculations indicate the 15x15 fuel exhibits slightly higher reactivity relative to 17x17 fuel for a reflected array of water moderated fuel at 100°F. As will be shown later, the reactivity difference between the 17x17 and 15x15 fuel lattices is small compared to the amount of margin obtained for calculations based on 17x17 fuel in the new fuel storage racks.

2.1 Model Data

The computer modeling of the new fuel 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 21) within the SCALE code system (Reference 22). The SCALE package automatically processes cross sections through the NITAWL and BONAMI codes 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.

The new fuel storage area at Surry (References 23 and 24) is common to Surry Units 1 and 2, and consists of nine parallel rows of storage racks with a total capacity of 126 fuel assemblies (Figure 1). Each storage location consists of a square 9-inch (inside measure) stainless steel box, 165 inches tall and with walls 1/8 inch thick. The storage area walls and floor are concrete. A steel grating at the top prevents accidental placement of a fuel assembly between storage cans. The storage area is normally dry.

Several fuel assembly and rack components were neglected in the model for simplicity and conservatism. The fuel assembly top and bottom nozzles (stainless steel), grids (Inconel and Zircaloy or ZIRLO), grid sleeves (stainless steel), and all storage rack structural materials other than the storage can itself were modeled as void or moderator regions. These omissions are conservative from a criticality standpoint because the materials, particularly the steel and Inconel components, are neutron absorbers. The original North Anna analysis also assumed the use of Zircaloy cladding and guide thimbles, while current Surry fuel uses ZIRLO for these components. The use of Zircaloy in the analysis is conservative for the ZIRLO assemblies, because the presence of niobium in the ZIRLO makes this advanced material a (marginally) stronger neutron absorber than Zircaloy.

The fuel assembly dimensions and material data used in the analysis are shown in Table 1. The corresponding information for the current Surry fuel design are also provided for comparison. A top view of the new fuel storage area is given in Figure 1, while Figure 2 shows the side view as modeled. Undesignated areas in these figures are air space under normal conditions.

2.2 Conditions Modeled

2.2.1 Normal Configuration

The base condition for the analysis consisted of a fully loaded storage area of 126 fresh assemblies, with nominal density (95% T.D.) fuel enriched to 4.3 weight percent U^{235}, and nominally centered in the storage cans. The air regions in the storage area were modeled as water vapor at a density of 10^{-8} g/cc.

2.2.2 Moderator Density Variation (Optimum Moderation)

Normal air humidity variations from dry conditions to heavy fog can result in water densities in the air ranging from 0 to 0.0025 g/cc (Reference 25). In addition, fire or a pipe break can result in flooding of the new fuel 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.

2.2.3 Fuel Assembly 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 inch. 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. 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 the fuel assembly pitch.

2.2.4 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 (i.e., the clad thickness was preserved).

2.3 Results of Criticality Calculations for the New Fuel Storage Racks

The K-effective for each rack condition analyzed is listed in Table 2. The results are also summarized below for each configuration.

2.3.1 Worst Case Normal Configuration

The base K-effective for the nominally-loaded 4.3 weight percent 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 a moderator density of 0.07 g/cc) results in an increase in K-effective of 0.009. A moderator density increase from 10^{-8} to 0.01 g/cc increases K-effective by 0.12. The worst case normal K-effective (excluding calculational uncertainty and bias) is therefore 0.572.

2.3.2 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 (optimum moderator case) at a water density of 0.07 g/cc, as shown in Figure 3. The K-effective increase associated with a water density increase from 0.01 g/cc to 0.07 g/cc is 0.312 ΔK. Note that at a moderator density of 0.998 g/cc, the K-effective is nearly identical to the peak value calculated at a moderator density of 0.07 g/cc.

Fuel compaction due to dropping fuel assemblies results in a gain of only 0.01 in the value of K-effective. (The value is 0.027 ΔK if the fuel height change is neglected in the calculation.) No accounting for pitch changes is 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 calculation of the normal configuration K-effective. The worst case abnormal configuration K-effective, exclusive of uncertainty or bias, is 0.884.

2.3.3 Effect of Calculational Uncertainty and Bias

The statistical uncertainty of the KENO new fuel storage rack calculations is less than ±0.004 ΔK 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 ΔK in the predicted versus experimental criticality. This bias is consistent with other reported values based on a larger number of critical experiments (Reference 26), although the minimum calculated K-effective in Reference 26 is lower by more than 1% (minimum K-effective = 0.974). Adding the average bias and uncertainty to the worst case values from Sections 2.3.1 and 2.3.2 above yields the following results:

Assuming the average bias:

Worst case normal configuration K-effective:	0.587 [= 0.572 + 0.004 + 0.011]
Worst case abnormal configuration K-effective:	0.899 [= 0.884 + 0.004 + 0.011]
K-effective limit for new fuel storage area:	0.980
Margin:	0.081 ΔK [= 0.980 - 0.899]

Assuming the worst case bias from Reference 26:

Worst case normal configuration K-effective:	0.602 [= 0.572 + 0.004 + 0.026]
Worst case abnormal configuration K-effective:	0.914 [= 0.884 + 0.004 + 0.026]
K-effective limit for new fuel storage area:	0.980
Margin:	0.066 ΔK [= 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 26 is applied.

2.3.4 Effect of Fuel Assembly Design

The 15x15 fuel assembly design used for Surry Units 1 and 2 contains slightly less UO_2 than the 17x17 fuel assembly design used for the above calculations, although the assemblies have comparable overall dimensions. Fuel type sensitivity calculations for criticality of the spent fuel storage racks using the PHOENIX-P code indicate that the 15x15 fuel exhibits a slightly higher reactivity (≈0.5% ΔK) relative to 17x17. The above-calculated K-effective values are therefore increased by 0.005 ΔK to account for this reactivity difference between the 15x15 and 17x17 fuel lattices.

In addition, the above analysis does not cover any fuel or design tolerance uncertainties, such as variation in as-built fuel density or fuel pellet dimensions, or variations in storage box thickness. The uncertainty calculated for the Surry spent fuel pool evaluation (0.01064 ΔK, from Section

3.3.3, below) is used as a conservative approximation of these tolerance uncertainties. The conservatism of this value arises from inclusion of some items in the calculation that are also explicitly considered in the calculation for the new fuel storage area, including the calculational uncertainty, methodology uncertainty, eccentric positioning tolerance, cell pitch tolerance, and fuel enrichment tolerance.

Increasing the above-calculated K-effective values by 0.01564 (rounded to 0.016) to account for the reactivity difference between the 15x15 and 17x17 fuel lattices and the tolerance uncertainties only slightly decreases the margins calculated above for the 17x17 fuel assembly design, as shown:

Assuming the average bias:

Worst case normal configuration K-effective:	0.603 [= 0.587 + 0.016]
Worst case abnormal configuration K-effective:	0.915 [= 0.899 + 0.016]
K-effective limit for new fuel storage area:	0.980
Margin:	0.065 ΔK [= 0.980 - 0.915]

Assuming the worst case bias from Reference 26:

Worst case normal configuration K-effective:	0.618 [= 0.602 + 0.016]
Worst case abnormal configuration K-effective:	0.930 [= 0.914 + 0.016]
K-effective limit for new fuel storage area:	0.980
Margin:	0.050 ΔK [= 0.980 - 0.930]

3. Impact on Spent Fuel Storage

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between fuel assemblies and inserting a neutron poison between them. The maximum allowable K-effective for the spent fuel racks is 0.95, per Section 9.1.2 of the Standard Review Plan (Reference 8).

The Surry spent fuel pool is common to Surry Units 1 and 2, and the existing spent fuel storage rack was previously qualified for storage of both 15 x 15 and 17 x 17 fuel assembly types with maximum enrichments up to 4.1 weight percent U^{235}. The Surry spent fuel pool is divided into two regions. Region 1 comprises the first three rows of storage racks (324 storage locations) adjacent to the fuel building trolley load block. This region is considered to be susceptible to damage from a dropped fuel storage cask. Any spent fuel assembly placed in Region 1 must have an average burnup greater than the current burnup versus initial enrichment curve, given as Figure 5.4-1 of the Surry Technical Specifications. Region 2 of the Surry spent fuel pool comprises the remainder of the storage racks (720 storage locations). Presently, Region 2 can contain fuel up to 4.1 weight percent U^{235} without any restrictions on assembly burnup.

The Surry spent fuel racks have been reanalyzed to allow an increase in the maximum enrichment to 4.3 weight percent U^{235}. The methodology used by the Virginia Electric and Power Company follows the Westinghouse Owner's Group (WOG) methodology, which is described in Reference 27 and was approved by the NRC in Reference 28. Differences between the methodology used for analysis of the Surry spent fuel racks and the WOG methodology are discussed in Section 3.2 below.

The analysis demonstrates that K-effective in the Surry spent fuel pool will be maintained ≤ 0.95, including uncertainties under normal and accident conditions with no soluble boron, except for accidents which assume the presence of soluble boron. Assuming soluble boron for these accidents is appropriate based on the "double contingency principle" of Reference 29.

3.1 Spent Fuel Pool Design Description

The Surry spent fuel storage rack layout is depicted in Figure 4. A top view of a spent fuel rack storage cell is shown in Figure 5, and Figure 6 shows the side view as modeled. The storage cell is constructed of type 304 stainless steel, and has no poison panels. The fuel parameters relevant to the analysis of the spent fuel storage racks are shown in Table 1. All past and present Surry and North Anna fuel designs were considered in this analysis. The cladding material (fuel and guide tube) for all fuel products is Zircaloy, except for the most recent fuel, which uses ZIRLO.

3.2 Analytical Methods for Evaluation of Spent Fuel Storage Racks

The computer modeling of the spent fuel storage racks was performed using the Monte Carlo program, KENO-V.a, that was also used for the analysis of the new fuel storage area (Section 2.1). The KENO-V.a code and cross sections were verified by comparison to critical experiment data for fuel assemblies similar to those for which the racks were designed. These benchmarking data are sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps, and soluble boron in the moderator. These benchmark cases consist of fifty-nine validation criticality test cases for UO_2 fuel lattices (Reference 30).

The design method which insures the subcriticality of the spent fuel storage racks uses the BONAMI and NITAWL-II codes for cross section generation and the KENO-V.a code for reactivity determination. The 238-group ENDF/B-V cross section library is the starting point for all cross sections used for the KENO-V.a benchmarks and KENO-V.a storage cell calculations. BONAMI performs a resonance self-shielding calculation based on the Bondarenko method, and produces problem dependent master data sets. NITAWL-II performs problem dependent resonance shielding calculations by applying the Nordheim Integral Treatment. These multigroup cross section sets are then used as input to KENO-V.a, which is a three dimensional Monte Carlo theory program designed for reactivity calculations. KENO-V.a calculations are always performed with sufficient neutron histories to assure convergence. The benchmark criticals and spent fuel pool criticality cases used at least 1,000,000 neutron histories.

The benchmark critical experiments resulted in an average KENO-V.a K-effective of 0.99643, which compared to a critical K-effective of 1.0 gives a KENO-V.a model bias of 0.00357 ΔK. The standard deviation of the bias value is 0.00049 ΔK. The 95/95 one-sided tolerance limit factor for the 59 values is 2.026 (Reference 31). Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity due to the method is not greater than 0.00099 ΔK.

The design method used to insure that a subcritical condition is maintained in the fuel storage rack is similar to the approved Westinghouse Owners Group (WOG) methodology, which is described in References 27 and 28, with the following exceptions.

1. The WOG methodology assumed a nominal UO_2 density of 95.0% T.D. The Surry spent fuel pool analysis used a nominal UO_2 density 95.5% T.D., based on recent as-built fuel assembly uranium-loading information.
2. The WOG methodology assumed a UO_2 density variation of ±2.0% T.D. about the nominal reference density, and variation in the fuel pellet dishing fraction from 0% to twice the nominal pellet dishing fraction. The analysis for the Surry spent fuel pool assumed a ±1.5% T.D. variation in fuel pellet density, based on Westinghouse manufacturing specifications and recent as-built fuel assembly uranium-loading information.
3. The WOG methodology used 227 group ENDF/B-V cross sections. These cross sections have not been made available to the Virginia Electric and Power Company by Oak Ridge National Laboratory. Therefore, the Surry spent fuel pool analysis used 238 group ENDF/B-V cross sections, which have been extensively benchmarked as discussed above.
4. The Surry spent fuel pool analysis does not require any boron credit. Therefore, the WOG boron credit methodology is not applicable.
5. The WOG methodology uses a nominal temperature of 68°F and pressure of 14.7 psia. In the analysis for the Surry spent fuel pool, the nominal temperature was set to the most conservative value over the typical temperature range, which was determined to be 170°F. The nominal pressure was set to 28 psia to account for the effects of the spent fuel pool water depth.
6. Although the NRC-approved WOG methodology does not require consideration of the effect of the axial burnup distribution on fuel assembly reactivity, an axial burnup gradient reactivity bias was applied to the evaluation of the Surry spent fuel pool.
7. The WOG methodology includes B^{10} self-shielding bias for spent fuel pools with poison panels. The Surry spent fuel storage cells do not include any poison panels, so this bias is not necessary in the Surry analysis. In addition, the Surry reactivity and tolerance calculations do not account for any poison panels.

Two different KENO-V.a models were used. One model represented an infinitely reflected single spent fuel storage cell, while the other model represented the entire Surry spent fuel storage pool.

The storage cells in the full pool model are placed in arrays to model each fuel storage rack, which are placed in their appropriate fuel pool location along with the fuel transfer canal and concrete buttress. This configuration is then surrounded with the stainless steel liner and concrete walls and floor.

Reactivity equivalencing and tolerance calculations were performed using the Westinghouse PHOENIX-P code. The NRC Safety Evaluation Report on the Westinghouse methodology (Reference 28) states that the benchmarking performed to support the Westinghouse methodology (Reference 27) covers the range of lattice parameters and configurations encompassing present fuel storage configurations as realistically as possible. Additionally, the PHOENIX-P isotopic comparisons to measurements detailed in the Westinghouse report were found to agree well for all measured isotopes throughout the burnup range. Based on the NRC acceptance of PHOENIX-P and their approval of the Westinghouse methodology described in Reference 27, further benchmarking was not performed for the Surry spent fuel pool analysis.

3.3 Criticality Analysis for Spent Fuel Storage Racks

This section describes the analytical techniques and models employed to perform the criticality analysis for the Surry spent fuel storage racks. Section 3.3.1 describes the reactivity calculations performed for the spent fuel storage racks using the defined nominal enrichments, storage configurations and rack conditions. Section 3.3.2 describes the tolerance calculations used to determine the reactivity uncertainty associated with fuel assembly and storage rack tolerances. Section 3.3.3 discusses the final 95 percent probability with a 95 percent confidence interval (95/95) K-effective calculations performed to ensure K-effective in the Surry spent fuel pool is less than or equal to 0.95.

3.3.1 Reactivity Calculations Using KENO-V.a

The KENO-V.a computer code was used to establish a nominal reference reactivity using fresh assemblies, which shows that storage of fuel assemblies in the Surry spent fuel storage racks meets the 0.95 K-effective criticality acceptance criteria.

A KENO-V.a model was developed for storage of fuel assemblies in the Surry spent fuel storage rack assuming no soluble boron. This model was based on the following assumptions.

1. The fuel assembly parameters relevant to the criticality analysis are based on the Surry 15 x 15 LOPAR (all Inconel grid) fuel design parameters shown in Table 1. Sensitivity cases showed that the Surry LOPAR fuel design is more reactive than the other Surry and North Anna fuel designs for the spent fuel pool conditions analyzed.
2. Each fuel assembly contains UO_2 at a nominal enrichment of 4.25 weight percent U^{235} over the entire length of each rod. When enrichment and manufacturing tolerances are considered, this is the maximum nominal enrichment which can be specified to ensure that the maximum fuel enrichment will not exceed 4.3 weight percent U^{235}.

3. The fuel pellets are modeled assuming nominal values for fuel density and dishing fraction.
4. No burnup credit is taken for any fuel assemblies.
5. No credit is taken for any U^{234} or U^{236} in the fuel.
6. No credit is taken for the presence of any spacer grids or spacer sleeves.
7. No credit is taken for any burnable absorber in the fuel rods or assemblies.
8. The moderator is water with 0 ppm soluble boron at a temperature of 170^0F at 28 psia. Sensitivity calculations determined that this temperature condition is the most reactive over the normal spent fuel pool temperature range.
9. All available storage cells are loaded with fuel assemblies.

KENO-V.a calculations were performed using both the infinitely reflected single storage cell model and the full spent fuel pool model. Comparison of the K-effective values determined for the two cases show that the single storage cell model slightly overpredicts the full spent fuel pool reactivity (by approximately 0.001 ΔK). Therefore, use of the single storage cell KENO-V.a model for this analysis is valid. The K-effective for the single storage cell model was 0.92950 ± 0.00079. No credit was taken for the conservatism present in the single storage cell model.

3.3.2 Tolerance Calculations Using PHOENIX-P

To evaluate the reactivity effects of possible variations in material characteristics and mechanical/construction dimensions, perturbation calculations were performed using Westinghouse's PHOENIX-P code. The material and mechanical uncertainties covered by these calculations include:

1. Enrichment tolerance of ±0.05 weight percent U^{235} about the nominal fresh reference enrichment of 4.25 weight percent U^{235}.
2. Variation of ±1.5% T. D. about the nominal reference UO_2 density.
3. Variation of 0% to twice the nominal (2.4%) fuel pellet dishing fraction.
4. Tolerance about the nominal reference storage cell inner dimension (±1/16 inch).
5. Tolerance about the nominal storage cell center-to-center pitch (±1/4 inch).
6. Tolerance about the nominal reference storage cell material thickness (±0.005 inch).
7. Asymmetric positioning of fuel assemblies within the storage cells (0.246 inch mispositioning).

The impact of each of these tolerances on the calculated K-effective is given in Table 3.

3.3.3 95/95 K-effective Calculations

To develop the 95 percent probability K-effective for the Surry spent fuel storage racks at a 95 percent confidence level, the following calculational and methodology biases were considered:

1. The benchmarking bias determined from comparison of KENO-V.a calculations to benchmark critical experiments. This bias is 0.00357 ΔK, as discussed in Section 3.2 above.
2. The reactivity bias associated with the normal range of spent fuel pool temperatures. Because all calculations for this analysis were performed at the moderator temperature conditions that give the most reactive condition, this bias is set to zero for the Surry analysis.

In addition, the uncertainty associated with material characteristics, mechanical construction, and the KENO-V.a methodology must be addressed. This uncertainty is determined by statistically combining the tolerances described in Section 3.3.2 above with the following uncertainties:

1. The 95/95 confidence level uncertainty on the KENO-V.a nominal reference K-effective (i.e., calculational uncertainty).
2. The 95/95 confidence level uncertainty in the KENO-V.a benchmarking bias from Section 3.2. This uncertainty is 0.00099 ΔK.

The combined uncertainty is 0.01064, as shown in Table 3. The following formula was then used to determine the 95/95 K-effective for the Surry spent fuel storage racks:

K-effective = Knominal + Bmethod + Btemp + Buncert

where:

Knominal = nominal conditions KENO-V.a K-effective (0.92950)
Bmethod = method bias determined from benchmark critical comparisons (0.00357)
Btemp = temperature bias (0.0)
Buncert = uncertainty associated with material characteristics, mechanical construction, and KENO-V.a method (0.01064, see Table 3)

Substituting the appropriate values, the resulting spent fuel pool K-effective is 0.94371. Since this K-effective is less than 0.95, the Surry spent fuel racks satisfy the requirement of NUREG-0800 (Reference 8) with a maximum fuel enrichment of 4.3 weight percent U^{235}.

3.4 Postulated Accidents for Spent Fuel Pool

Accident conditions must also be addressed in the spent fuel criticality analysis to ensure that K-effective is maintained less than or equal to 0.95. Consistent with the methodology described in Reference 27, two types of accidents were considered which can cause reactivity to increase in the spent fuel rack. The first accident type, a fuel assembly misplacement, involves the

placement of a fuel assembly into a position for which any restrictions on location, enrichment, or burnup are not satisfied. The second accident type, a pool water temperature change, involves an increase or decrease in the spent fuel pool water temperature and density.

For an occurrence of either type of the postulated accidents, the double contingency principle of ANSI/ANS 8.1-1983 (Reference 29) can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, the presence of soluble boron in the spent fuel pool storage water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event (i.e. a boron dilution accident).

The evaluation presented in Section 3.3 applies to a spent fuel storage rack configuration containing all fresh fuel at the maximum permissible proposed Surry fuel enrichment, with no restrictions on assembly location. Therefore, the first postulated accident that would increase spent fuel storage rack reactivity is already bounded by the previously described base (non-accident) analysis.

Two additional fuel assembly misplacement accidents that were considered were determined to have no impact on reactivity. These accidents include fuel assembly drop on top of a rack and fuel assembly drop between rack modules or between rack modules and the spent fuel pool wall. For the first scenario (fuel assembly drop on top of a rack), the rack structure pertinent for criticality control is not excessively deformed by the fuel assembly. KENO-V.a sensitivity cases confirmed that a dropped assembly which comes to rest either horizontally or vertically on top of the rack has sufficient water separating it from the active fuel height of stored assemblies to preclude neutron interaction. Additionally, PHOENIX-P sensitivity cases showed that placing an assembly outside of the racks per the second accident scenario is bounded by the normal conditions analysis in Section 3.3 above.

Pool temperature sensitivity cases were performed using the PHOENIX-P computer code. The temperatures in these cases ranged from 50°F to 246.4°F, the temperature at which boiling would be expected to occur in the spent fuel pool. The normal conditions analysis (Section 3.3) covered a normal temperature range from 50°F to 170°F, and showed that K-effective is less than 0.95 with no boron present in the spent fuel pool. It was determined that an increase in pool temperature from 170°F to 246.4°F increases the spent fuel pool K-effective less than the worth of the boron normally present in the spent fuel pool. Therefore, the 0.95 K-effective limit would be met for a pool water temperature increase.

For Surry, a fuel storage cask handling accident scenario must also be considered. This accident assumes a fuel storage cask rotates and falls against the fuel storage racks next to the cask loading area. To evaluate this accident, the Surry spent fuel pool is divided into two regions. Region 1 comprises the first three rows of fuel storage racks (324 locations) adjacent to the fuel building trolley load block. This region is susceptible to the storage cask handling accident. Region 2 comprises the remainder of the Surry spent fuel storage pool.

To evaluate the impact of a storage cask handling accident on spent fuel pool criticality, the deformed fuel and the associated storage racks were assumed to be at the optimum pitch. Calculations were performed using KENO-V.a to determine the maximum fresh fuel enrichment under the optimum pitch assumption that meets the 0.95 K-effective limit, including all applicable uncertainties and tolerances (see Table 4). The methodology employed for the 95/95 K-effective calculations in Section 3.3.3 was used for these calculations. Note that some uncertainty components were applied as biases for this accident as indicated in Table 4. The results of these cases demonstrate that fresh fuel at an initial enrichment of 1.9 weight percent U^{235} results in a Region 1 spent fuel pool K-effective of 0.93204, which is less than the 0.95 K-effective criteria. It is therefore concluded that any fuel with an initial U^{235} enrichment less or equal to 1.9 weight percent may be loaded into Region 1 of the Surry spent fuel pool.

To allow the loading of fuel with higher initial enrichments in Region 1 of the Surry spent fuel pool, credit must be taken for the fuel burnup. For this calculation, reactivity equivalencing calculations were performed using the PHOENIX-P code and following the methodology outlined in Reference 27. A series of reactivity calculations were performed to generate a set of fuel assembly initial enrichment-discharge burnup ordered pairs, which all yield equivalent K-effective values for the Surry spent fuel storage racks. These burnup credit cases incorporated the following conservatisms and uncertainties:

1. Fuel depletions were performed at a conservatively high boron concentration to enhance the predicted buildup of plutonium. This conservatively makes the fuel assembly more reactive when stored in the spent fuel storage racks.
2. Burnup credit cases assumed no xenon, thus maximizing the assembly reactivity.
3. A PHOENIX-P code uncertainty was applied. This uncertainty starts at zero for zero burnup and increases linearly with burnup, passing through 0.01 ΔK at a burnup of 30,000 MWD/MTU.
4. An axial burnup gradient reactivity bias was applied. This bias starts at zero for zero burnup and increases linearly with burnup, passing through 0.02 ΔK at a burnup of 36,250 MWD/MTU.
5. A reactivity bias was applied to account for changes in optimum pitch due to burnup and enrichment changes. This bias was determined through PHOENIX-P sensitivity calculations to be zero at zero burnup, increasing linearly with enrichment to 0.02893 ΔK at 5.0 weight percent U^{235}.
6. An uncertainty of 5% was also applied to assembly burnup to reflect uncertainties in burnup measurements.

Figure 7 shows the results of the burnup credit reactivity equivalencing calculations. Use of fuel with a burnup and enrichment combination which falls above the Figure 7 curve in Region 1 of the Surry spent fuel pool would ensure that the spent fuel pool K-effective remains less than or equal to 0.95 during a fuel storage cask handling accident. Therefore, fuel which falls within the acceptable area of Figure 7 is acceptable for storage in Region 1 of the Surry spent fuel storage pool.

4. Other Considerations

4.1 Safety Analysis Inputs

Use of a higher initial fuel enrichment would be expected to slightly increase the fuel temperatures and rod internal pressures used as input to LOCA analyses and other safety calculations. However, the fuel temperatures and fuel rod internal pressures used in Surry safety and LOCA analyses are generic values that are valid for fuel enrichments up to 5 weight percent U^{235}. The proposed increase in maximum fuel enrichment from 4.1 to 4.3 weight percent U^{235} will therefore not affect these inputs to any Surry safety analysis.

4.2 Radiological Consequences of Accidents

Evaluations of the consequences of accident scenarios described in the Surry UFSAR will not be affected by the increase in the initial fuel enrichment. Assessments by the NRC and fuel suppliers of the impact of increasing fuel burnup from 30,000 MWD/MTU to 60,000 MWD/MTU (References 4 and 5) demonstrated that the radionuclide concentrations in the fuel are largely a function of the reactor power and fuel integral power history rather than fuel enrichment. The operating power of the Surry units will not be changing as a result of the proposed increase in maximum initial fuel enrichment.

For most accident scenarios described in the UFSAR, the primary contributions to doses are from short-lived iodine, xenon and krypton isotopes present in the pellet to clad gap during operation or, for the fuel handling accident, shortly after shutdown. The quantities of these isotopes present in the fuel are a function of the number of fissions which have occurred (burnup) rather than the number of fissionable atoms initially present in the fuel (enrichment). At higher burnups, the levels of the isotopes of interest for most accident scenarios tend to reach an equilibrium condition between production and decay. As the fuel at Surry Units 1 and 2 will continue to be limited to a lead rod burnup of 60,000 MWD/MTU (References 6 and 7), the increase in the initial fuel enrichment limit from 4.1 to 4.3 weight percent U^{235} will not affect the source terms used to evaluate the consequences of postulated accident conditions.

A storage cask handling accident differs from this characterization because isotopic decay during the residence time in the spent fuel pool affects the nature of the source term for the analysis. Surry Technical Specifications require that during spent fuel cask handling, the fuel in the portion of the pool that may be affected by a cask drop accident (Region 1) must:

(a) satisfy the burnup versus initial enrichment curve in Figure 5.4-1 of the Technical Specifications, and
(b) have decayed for at least 150 days after discharge.

As discussed in Section 3 above, with the increase of the maximum initial fuel enrichment to 4.3 weight percent U^{235}, Figure 5.4-1 of the Technical Specifications will be modified as shown in Figure 7 to ensure that criticality limits are satisfied for placement of the higher enrichment fuel

in Region 1 of the spent fuel pool. The minimum decay time of 150 days is retained, which means that for a storage cask handling accident most of the shorter-lived isotopes normally considered in dose calculations, such as most iodines, will have dropped to very low levels. Consequently, longer-lived isotopes that accumulate with burnup and are normally very minor contributors to dose calculations - such as Kr^{85} - constitute a significant portion of the source term for such an analysis, even though the absolute quantities of these isotopes continue to be very small. However, the maximum burnup of the fuel to be stored in Region 1 of the Surry spent fuel pool will continue to be limited to a lead rod burnup of 60,000 MWD/MTU. The quantities of long-lived isotopes will therefore not increase above levels that are currently allowable for fuel in Region 1 of the spent fuel pool during cask handling operations. It is concluded that an increase in the maximum initial fuel enrichment from 4.1 to 4.3 weight percent U^{235} will not affect the environmental consequences of the storage cask drop accident for the Surry spent fuel pool.

4.3 Safety Analysis Limits

Other than the proposed changes to those Technical Specifications which define the maximum fuel enrichment, there are no changes to any Technical Specifications Limiting Conditions for Operation, operating or safety-related setpoints, or Technical Specifications basis associated with this enrichment increase. No changes are being made to the core power level, operating temperature or pressure, or any peaking factors. The existing limits on the primary system coolant activity will also remain applicable.

It may be noted that discussions have been continuing within the industry regarding reactivity insertion accidents and the potential for fuel melt to occur in high burnup fuel under such accident scenarios. However, the subject of these discussions is again related to the effects of the burnup achieved by the fuel on the properties of the material, rather than to the amount of U^{235} initially present in the fuel. No change in the lead rod burnup limit of the Surry fuel is requested with this Technical Specifications change request, and any resolution of outstanding questions in this area will be unaffected by the small increase in the initial enrichment to 4.3 weight percent U^{235}.

4.4 Impact on Reactor Vessel Fluence

The major factors affecting the reactor vessel fluence are the overall power level and the fuel peaking factors, particularly the radial peaking factors. These parameters are not being changed at this time, so use of fuel with a higher initial enrichment will not impact the vessel fluence. In addition, the impact on vessel fluence is checked as part of each reload safety evaluation. Therefore, if the use of a higher fuel enrichment indirectly leads to conditions (e.g., higher radial power distributions) that could adversely affect the vessel fluence, the change will be detected as part of the reload evaluation and incorporated into ongoing tracking of the fluence to the vessel.

4.5 Control Rod Insertion

Concerns have been raised on the possible effects of fuel burnup on the fuel assembly structure, potentially leading to conditions that could interfere with control rod insertion and thus safe shutdown of the reactor. It has been postulated that such conditions could develop at higher fuel burnups as the result of the mechanical design of the fuel assembly guide thimble tubes, the material used to fabricate the fuel assembly skeleton, and irradiation induced changes to the fuel assembly such as growth and corrosion. No difficulties with control rod insertion have been observed in the 15x15 fuel design used for Surry Units 1 and 2, and there are no changes to the mechanical design of the fuel assembly associated with this fuel enrichment increase. The irradiation induced factors which affect the skeleton in a manner that could conceivably ultimately affect the guide thimble geometry and thus the ability to insert the control rods as designed are burnup, rather than enrichment, related. Use of a slightly higher initial U^{235} enrichment in Surry fuel, with no increase in fuel burnup limits, will not adversely affect fuel performance in any way which would affect the ability of the control rods to fully insert.

Although incomplete control rod insertion has not been observed in our units, Virginia Electric and Power Company has continued to perform cycle specific calculations to verify that our core designs incorporate sufficient shutdown margin to ensure safe shutdown if some RCCAs in high burnup fuel fail to fully insert. An increase in the initial fuel enrichment from 4.1 to 4.3 weight percent of U^{235} will not affect the nature of such shutdown margin calculations for Surry Units 1 and 2.

4.6 Decay Heat Load for Full Core Offload

An increase in the maximum initial fuel enrichment from 4.1 to 4.3 weight percent U^{235} alone will not affect the decay heat load in the spent fuel pool during a full core offload. This type of calculation is primarily sensitive to parameters such as assumed operational power and burnup, and will not be significantly affected by a small increase in the initial fuel enrichment or even a small $F\Delta H$ increase. A reassessment of the decay heat load will be required if changes in fuel management are implemented which significantly affect the basic assumptions of the evaluation, such as the burnup of the discharged fuel, the assumed operating power prior to discharge, or the duration of the refueling outages.

4.7 Dry Fuel Storage

There is no fuel enrichment limit specified in the Surry Independent Spent Fuel Storage Installation (ISFSI) site license. Limits on the initial fuel enrichment are included as part of the licensing basis for each individual type of dry storage cask used at the Surry ISFSI. It should be noted that the cask designs currently in use at Surry are licensed for lower initial fuel enrichments and fuel assembly burnups than may be found in current generation fuel. Virginia Electric and Power Company recognizes that further evaluations and license amendments will be required to place higher enrichment fuel into dry storage casks. The evaluations performed to support such requests for amendments to the cask licenses will consider the appropriate fuel

design parameters, including the initial U^{235} enrichment and the burnup of the fuel assemblies to be placed into storage.

ASSESSMENT OF UNREVIEWED SAFETY QUESTION

The use of fuel with a higher initial U^{235} enrichment will not result in an unreviewed safety question as defined by 10 CFR 50.59. The basis for this determination is summarized below.

Probability and Consequences of Previously Evaluated Accidents

The proposed increase in maximum fuel enrichment will not increase the probability of an accident previously evaluated in the Surry Units 1 and 2 UFSAR. The only accidents for which the probability of occurrence is potentially affected by the fuel enrichment involve criticality events during fuel handling and storage. Criticality safety analyses have been performed that demonstrate that the K-effective during the handling and storage of both new and spent fuel is 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 other accidents previously evaluated in the Surry Units 1 and 2 UFSAR will not be increased.

Possibility of Accidents Not Previously Evaluated

The possibility of an accident which is different from any already discussed in the Surry Units 1 and 2 UFSAR is not created. Fuel with the higher initial enrichment will meet all applicable design criteria and will operate within existing Technical Specifications limits. Adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. The only potential impact of increased enrichment on fuel storage and handling involves the potential for criticality, which was addressed above. Existing safety analyses of record will remain applicable for use of fuel with the higher initial enrichment.

Probability and Consequences of Previously Evaluated Malfunction of Equipment Important to Safety

The probability of a malfunction of equipment important to safety previously evaluated in the Surry Units 1 and 2 UFSAR is not increased. The design of cores which incorporate fuel at the higher initial enrichment will meet all applicable design criteria. Adherence to applicable standards and acceptance criteria, including existing limits on fuel burnup, precludes new challenges to components and systems that could increase the probability of any previously evaluated malfunction of equipment important to safety. The use of a higher maximum fuel enrichment will not impose new performance requirements on any system or component such that any design criteria for fuel operation or storage will be exceeded. No new modes or limiting single failures are created by the use of a higher fuel enrichment. Safety analyses for the fuel

storage area have demonstrated that subcriticality will be maintained during fuel handling and storage.

Possibility of Malfunction of Equipment Important to Safety Not Previously Evaluated

The possibility of a malfunction of equipment important to safety different from any already evaluated in the Surry Units 1 and 2 UFSAR is not created. The design for Surry cycles which incorporate the higher enriched fuel will meet all applicable design criteria. Adherence to existing Technical Specifications and design limits will preclude new challenges to components and systems that could introduce a new type of malfunction of equipment important to safety. No new failure modes have been created for any system, component, or piece of equipment, and no new single failure mechanisms have been introduced. The use of higher enriched fuel has the potential to affect only criticality events during fuel handling and storage. Safety analyses demonstrated that K-effective will remain sufficiently low to ensure subcriticality, so no new releases will result and there is no impact on radiological consequences of accidents.

Margin of Safety

The margin of safety as defined in the Bases to any Surry Technical Specification is not reduced. Safety analyses of record will remain applicable for the operation of fuel with a higher initial U^{235} enrichment. Criticality analyses demonstrate that the limits on K-effective for the new and spent fuel storage areas will be satisfied. Therefore, there is adequate margin to ensure subcriticality during the storage and handling of fuel, and the requirements of 10 CFR 50 Appendix A General Design Criterion 62 are satisfied.

The Surry Units 1 and 2 Technical Specifications (Reference 32) ensure that the plants operate in a manner that provides acceptable levels of protection for the health and safety of the public. The Technical Specifications are based upon assumptions made in the safety and accident analyses, including those relating to the fuel enrichment and the design of the fuel storage areas. The Surry safety analyses for core operation will remain applicable for cores which use fuel with the higher U^{235} enrichment, and analyses have demonstrated that subcriticality will be ensured during fuel storage and handling accident scenarios. Therefore the regulated margin of safety as defined in the Technical Specifications is not affected by the proposed increase in initial fuel enrichment.

Based on the evaluations and analysis results presented in the foregoing safety significance evaluation, it has been demonstrated that increasing the Surry Units 1 and 2 maximum initial fuel enrichment to 4.3 weight percent U^{235} will not result in the acceptable safety limits for any incident being exceeded, or in any unreviewed safety questions as defined in 10 CFR 50.59.

SUMMARY AND CONCLUSIONS

Virginia Electric and Power Company proposes to increase the maximum fuel enrichment for Surry Units 1 and 2 from the current Technical Specifications limit of 4.1 weight percent U^{235}, to

4.3 weight percent U^{235}. The use of a higher fuel enrichment has the potential to impact criticality safety analyses for fuel handling and storage. Calculations were performed to address the impact on the new and spent fuel storage areas, which are common to Surry Units 1 and 2.

The analysis for the new fuel storage area was originally performed for North Anna Units 1 and 2, which have the same design, and for which the NRC has already approved a similar enrichment increase. Additional uncertainties were applied to account for the reactivity differences between the Surry (15x15) and North Anna (17x17) fuel lattices, and to address fuel and storage area design tolerances. It was demonstrated that the Surry new fuel storage area meets the criticality limit of K-effective < 0.98, and is safe under the criticality specifications set forth in Section 9.1.1 of the Standard Review Plan (NUREG-0800).

The Surry spent fuel racks were reanalyzed for a maximum enrichment of 4.3 weight percent U^{235}. The methodology used for this analysis is similar to the NRC-approved Westinghouse Owners Group Methodology (Reference 27). It was demonstrated that the K-effective in the Surry spent fuel pool will be maintained ≤ 0.95, including uncertainties, under normal and accident conditions with no soluble boron, except for those accidents which assume the presence of soluble boron. The Surry spent fuel pool will therefore remain safe under the criticality specifications set forth in Section 9.1.2 of the Standard Review Plan for the proposed fuel enrichment increase.

Use of a slightly higher initial fuel enrichment in Surry fuel assemblies will not affect the inputs to any Surry safety analysis, nor will the consequences of accident scenarios described in the Surry UFSAR be affected. Other areas which might be affected by a change in the fuel enrichment were reviewed, and no adverse impacts were identified. It was concluded that the use of fuel with a slightly higher initial U^{235} enrichment will not result in an unreviewed safety question as defined by 10 CFR 50.59.

Table 1
Fuel Assembly Data for
New Fuel Storage Area and
Spent Fuel Storage Rack Calculations

Design Feature	North Anna (17x17) Fuel Design	Current Surry (15x15) Fuel Design
Maximum Fuel Enrichment, w/o U^{235}	4.3	4.1
Assembly Pitch, inches	8.466	8.466
Fuel Rods per Assembly	264	204
Fuel Rod Pitch, inch	0.496	0.563
Clad Material	Zircaloy-4 or ZIRLO	Zircaloy-4 or ZIRLO
Clad Outer Diameter, inch	0.374	0.422
Fuel Clad Thickness, inch	0.0225	0.0243
Active Fuel Length, inches	144	144
Fuel Pellet Material	UO_2	UO_2
Fuel Pellet Outer Diameter, inch	0.3225	0.3659
Fuel Pellet to Clad Diametral Gap, inch	0.0065	0.0075
Pellet Density, % T.D.	95.0 (nominal) 95.5 (current typical)	95.0 (nominal) 95.5 (current typical)
Fuel Pellet Dishing Fraction, %	1.2	1.2
Guide Tubes per Assembly*	25	21
Guide Tube Material*	Zircaloy-4 or ZIRLO	Zircaloy-4 or ZIRLO
Guide Tube Outer Diameter, inch* - Inconel mixing vane grids (LOPAR fuel) - Zircaloy or ZIRLO mixing vane grids (SIF fuel)	 0.482 0.474	 0.546 0.533
Guide Tube Thickness, inch*	0.016	0.017

* Includes instrumentation tube. Diameters shown are in the upper portion of the guide thimbles, above the dashpot region.

Table 2
New Fuel Storage Rack K-Effective
Based on 4.3 weight percent U^{235} and 17x17 Fuel Array

Case Type	Moderator Density*	K-effective	Comments
Base	10^{-8}	0.44320 ± 0.00108	Nominal Base Case
Alt. Enrich	10^{-8}	0.43735 ± 0.00106	4.1 w/o U^{235} 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 ± 0.00168	
Pitch (Base)	10^{-8}	0.44265 ± 0.00114	Simplified for pitch change
Pitch - 0.5 inch	10^{-8}	0.44255 ± 0.00103	
Pitch - 1.0 inch	10^{-8}	0.44155 ± 0.00110	

*Note: Moderator density is KENO volume fraction. Nominal density is 0.9982 g/cc.

Table 2 (cont.)
New Fuel Storage Rack K-Effective
Based on 4.3 weight percent U^{235} and 17x17 Fuel Array

Case Type	Moderator Density*	K-effective	Comments
Pitch (Base)	0.07	0.87200 ± 0.00151	Simplified for pitch change
Pitch - 0.5 inch	0.07	0.88048 ± 0.00139	
Pitch - 1.0 inch	0.07	0.88139 ± 0.00153	
Dropped Fuel	0.065	0.89166 ± 0.00144	144 inch fuel height
Dropped Fuel	0.065	0.86610 ± 0.00148	119 inch fuel height
Dropped Fuel	10^{-8}	0.47018 ± 0.00102	144 inch fuel height
Dropped Fuel	10^{-8}	0.45327 ± 0.00112	119 inch fuel height

*Note: Moderator density is KENO volume fraction. Nominal density is 0.9982 g/cc.

Table 3
Spent Fuel Pool Uncertainties

Tolerance	ΔK
Enrichment (+0.05 weight percent)	0.00247
Density (+1.5% TD)	0.00284
Dishing Fraction (0%)	0.00234
Cell Pitch (-1/4 in)	0.00823
Cell Wall Thickness (-0.005 in)	0.00196
Cell I.D. (-1/16 in)	0.00005
Assembly Position	0.00439
Calculational Uncertainty	0.00130
Methodology Uncertainty	0.00099
Total Uncertainty (Buncert)	0.01064

The total uncertainty (Buncert) was determined by statistically summing each uncertainty component.

$$Buncert = \sqrt{\sum_i Unc_i^2}$$

Table 4
Spent Fuel Pool
Cask Handling Accident Uncertainties

Tolerance	ΔK
Enrichment (+0.05 weight percent)[1]	0.00000
Density (+1.5% TD)[1]	0.00000
Dishing Fraction (0%)	0.00102
Cell Pitch (-1/4 in)[2]	0.00000
Cell Wall Thickness (-0.005 in)[2]	0.00000
Cell I.D. (-1/16 in)[2]	0.00000
Assembly Position[2]	0.00000
Calculational Uncertainty	0.00081
Methodology Uncertainty	0.00099
Total Uncertainty (Buncert)	0.00164

The total uncertainty (Buncert) was determined by statistically summing each uncertainty component.

$$Buncert = \sqrt{\sum_i Unc_i^2}$$

Notes: [1] This component was included as a bias in the KENO-V.a calculations.
[2] This tolerance component was set to 0.0 because of the optimum pitch assumption.

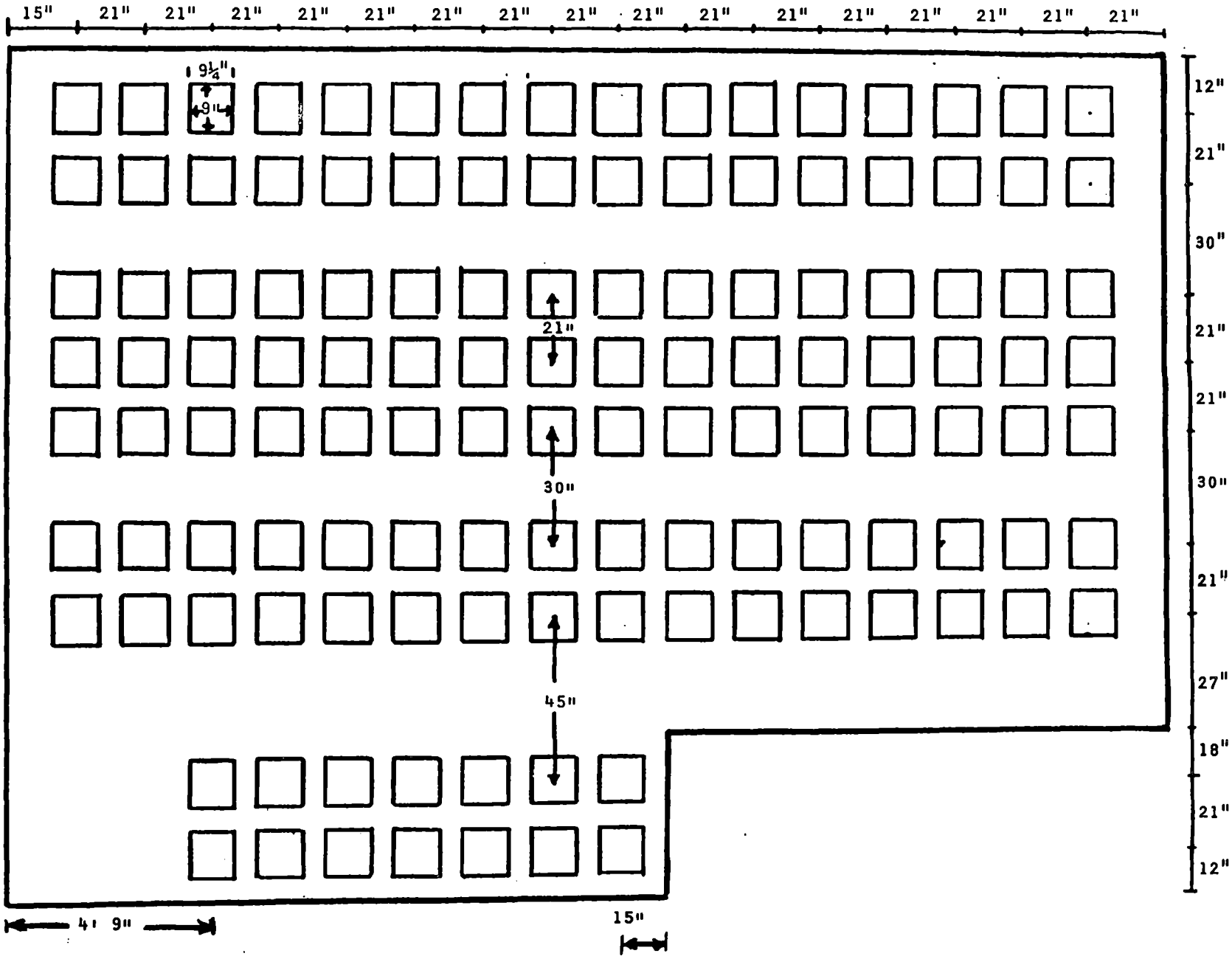


Figure 1.
New Fuel Storage Area (Top View)

Figure 2.
New Fuel Storage Area, Side View
(As Modeled)

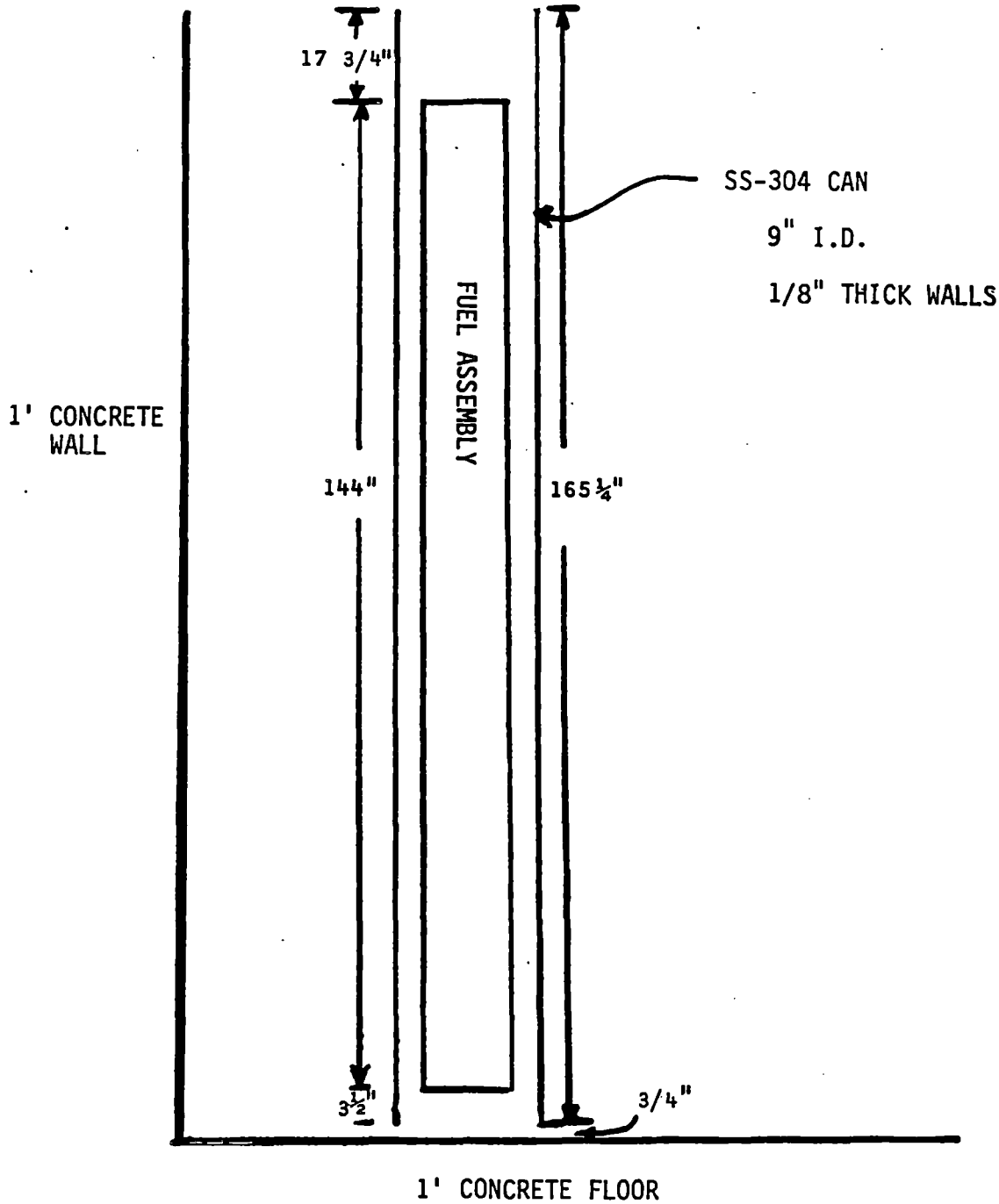


Figure 3.
K-Effective versus Moderator Density
for New Fuel Storage

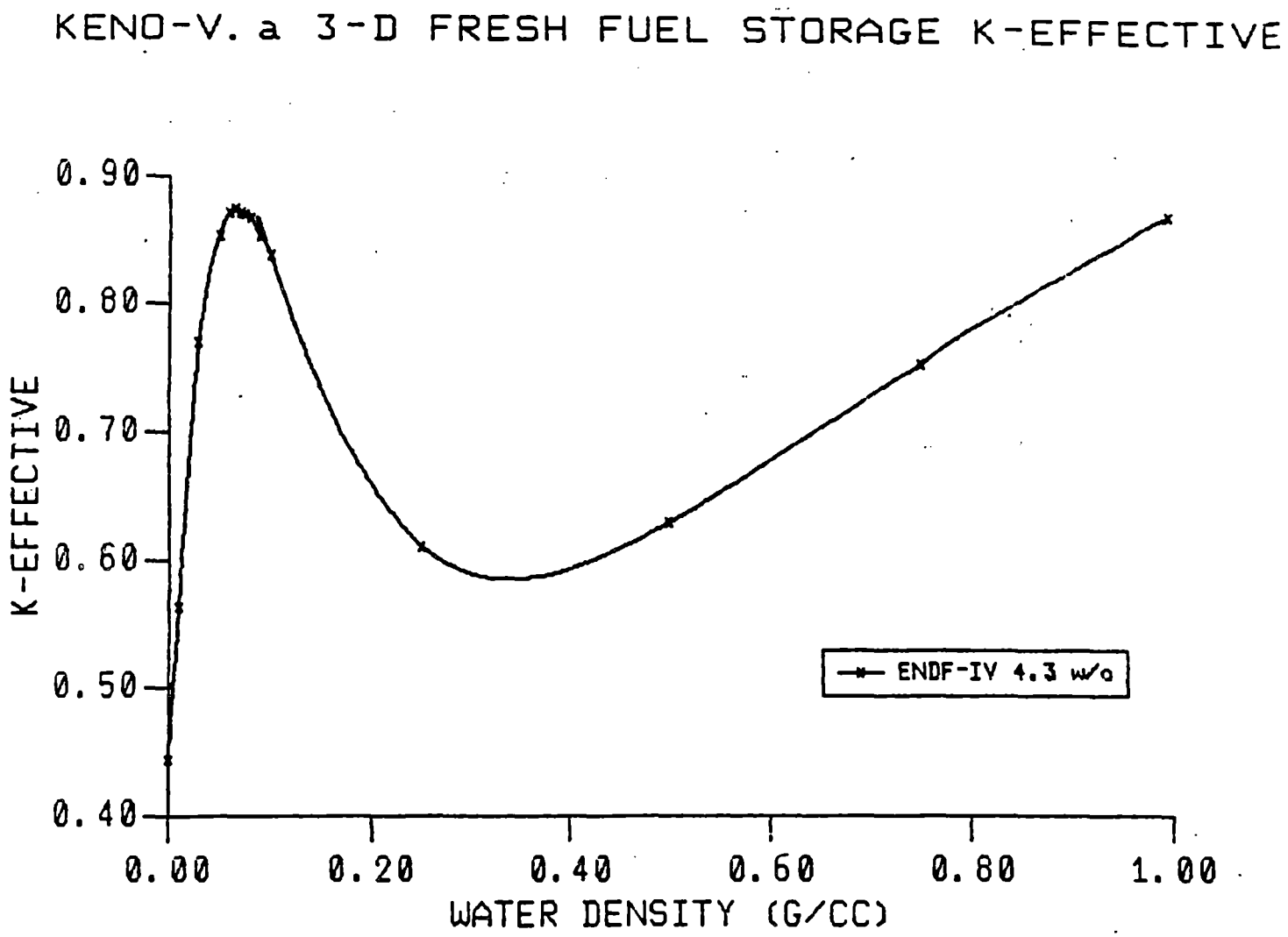
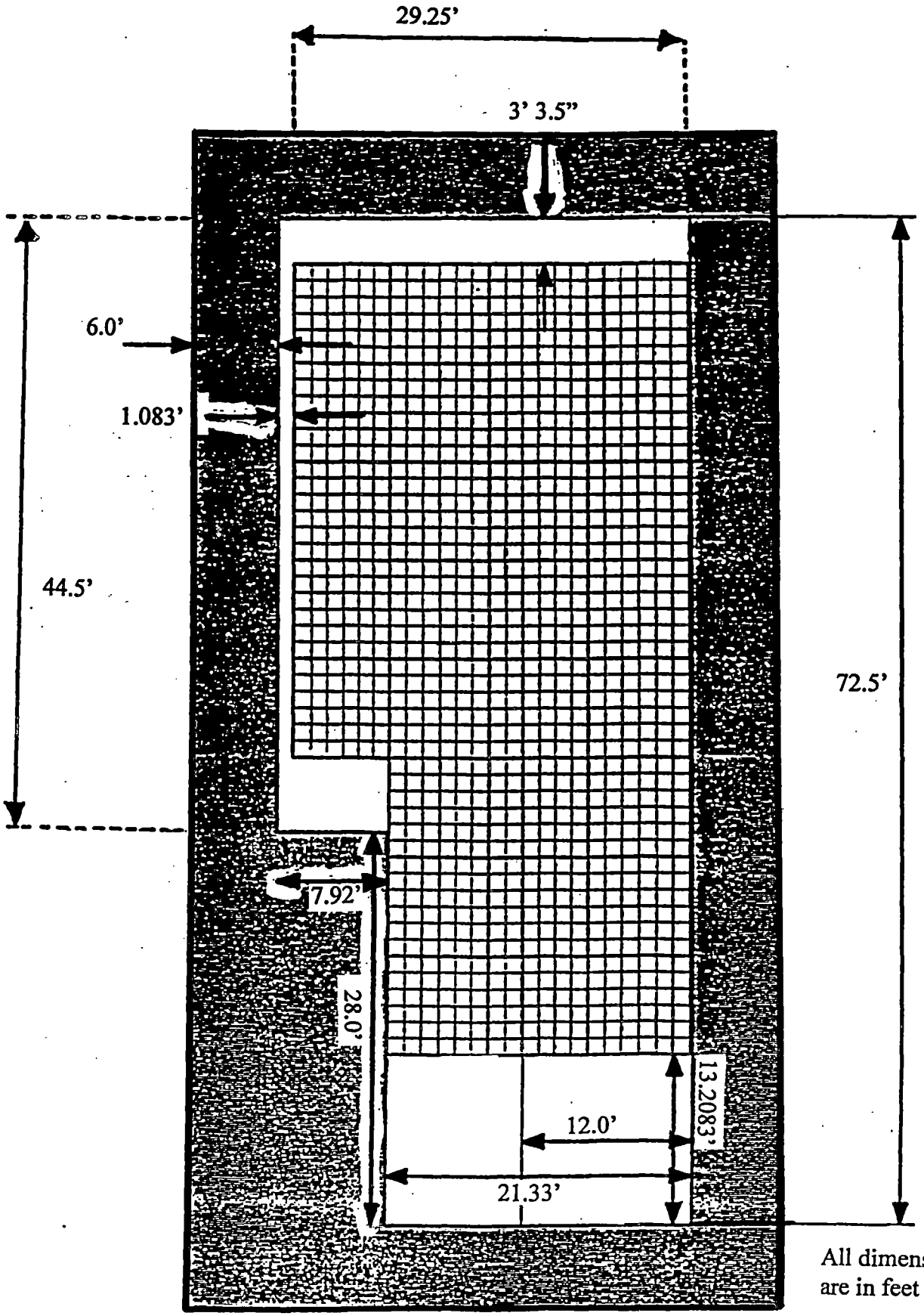


Figure 4
Surry Spent Fuel Storage Rack Arrangement



All dimensions are in feet

Figure 5
Surry Spent Fuel Storage Rack
Storage Cell Arrangement
(Not to Scale)

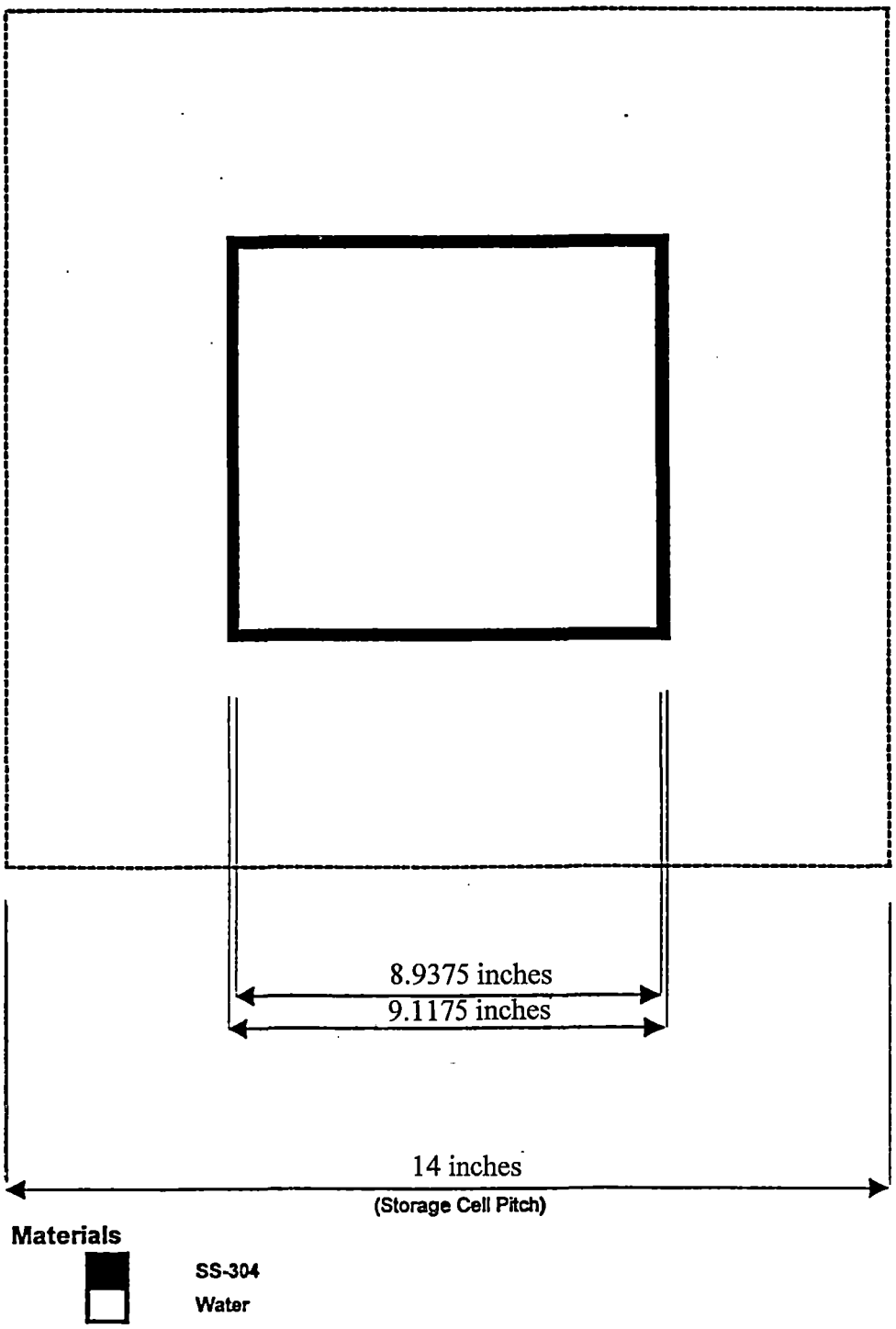


Figure 6
Surry Spent Fuel Storage Rack
Storage Cell Arrangement, Side View
(As Modeled)

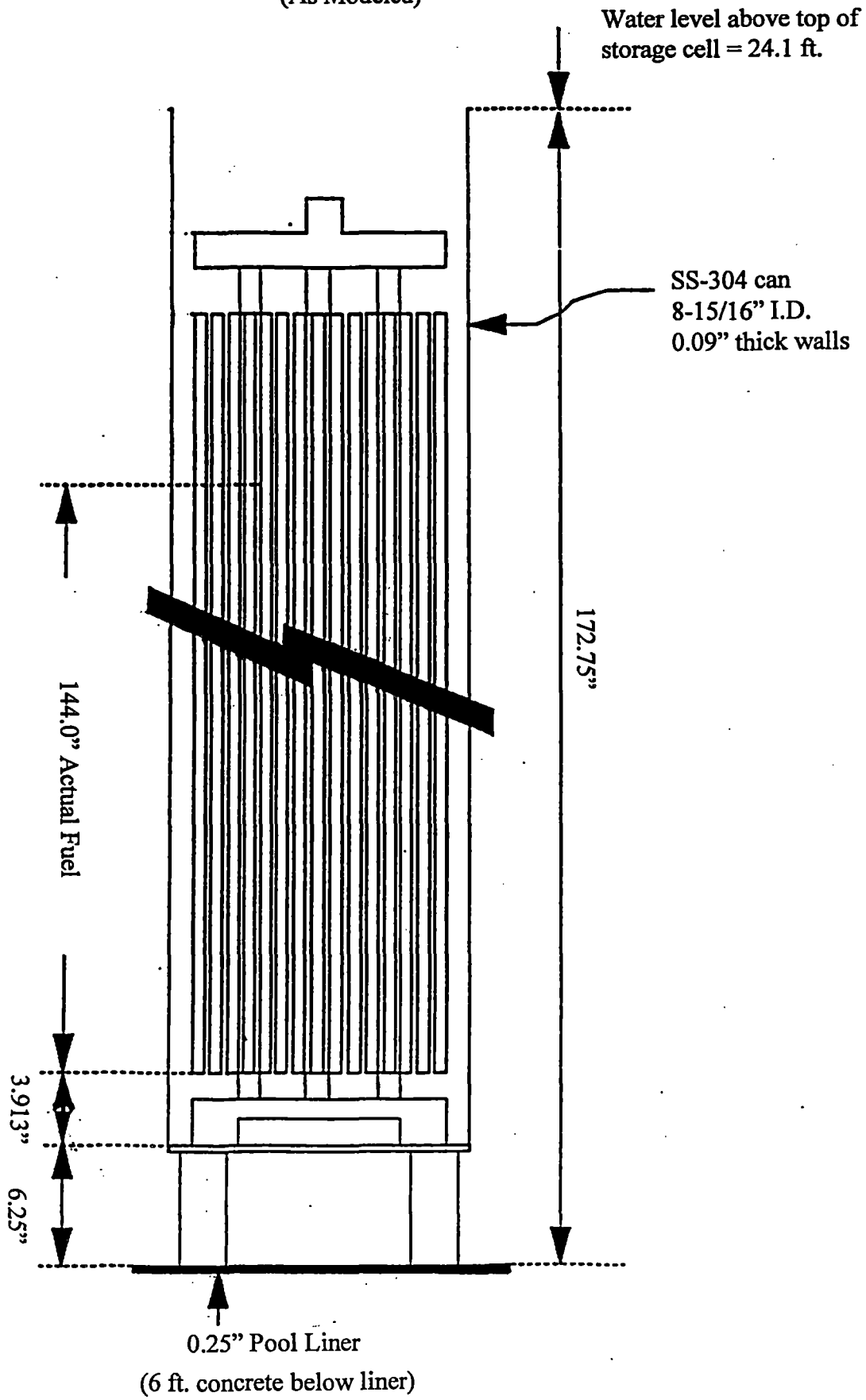
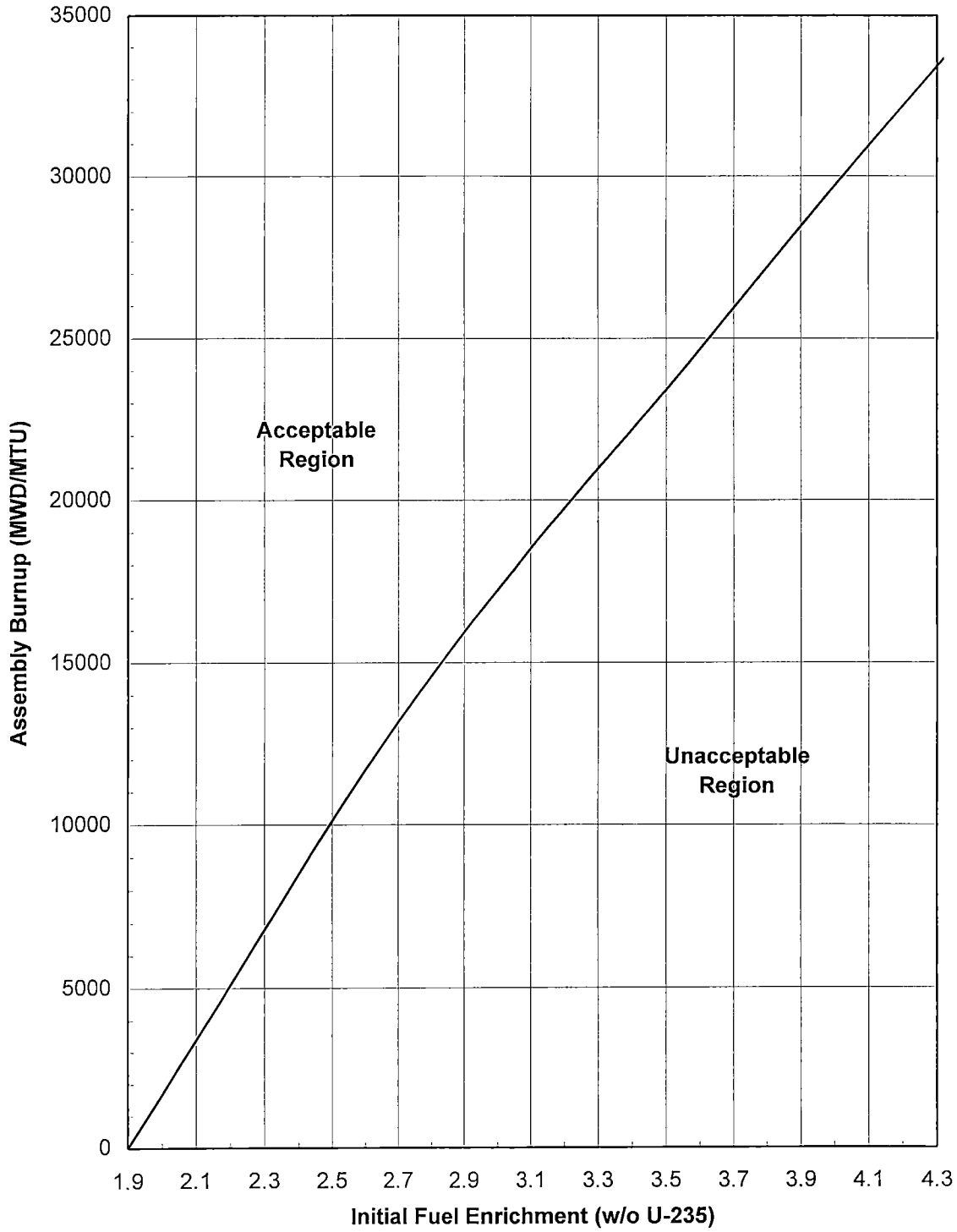


Figure 7
Surry Spent Fuel Pool
Region 1 Burnup Credit



REFERENCES

1. Letter from Leon B. Engle (U.S. Nuclear Regulatory Commission) to W. L. Stewart (Virginia Electric and Power Company), "North Anna Units 1 and 2 - Issuance of Amendments Re: Fuel Enrichment (TAC Nos. 69798 and 69799)," June 6, 1990.

2. Letter from Joseph D. Neighbors (U.S. Nuclear Regulatory Commission) to R. H. Leasburg (Virginia Electric and Power Company) issuing Amendments 73 and 74 to the Surry Power Station Units 1 and 2 Facility Operating Licenses, January 19, 1982.

3. Letter from Steven A. Varga (U.S. Nuclear Regulatory Commission) to W. L. Stewart (Virginia Electric and Power Company) increasing restriction on batch average burnup to 45,000 MWD/MTU, April 9, 1984.

4. S. L. Davidson and W. L. Kramer, Editors, "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A (Proprietary), December, 1985.

5. D. A. Baker, et al., "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," NUREG/CR-5009, PNL-6258, February 1988.

6. Letter from B. C. Buckley and L. B. Engle (U.S. Nuclear Regulatory Commission) to W. L. Stewart (Virginia Electric and Power Company), "Surry, Units 1 and 2, and North Anna, Units 1 and 2 - Removal of 45,000 MWD/MTU Batch Average Burnup Restriction (TAC Nos. M87767, M87768, M87812, and M87813)," December 14, 1993.

7. Letter from B. C. Buckley and L. B. Engle (U.S. Nuclear Regulatory Commission) to W. L. Stewart (Virginia Electric and Power Company), "Surry, Units 1 and 2, and North Anna, Units 1 and 2 - Removal of 45,000 MWD/MTU Batch Average Burnup Restriction (TAC Nos. M87767, M87768, M87812, and M87813)," April 29, 1994.

8. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Standard Review Plan," NUREG-0800, Section 9.1.1, 'New Fuel Storage,' Revision 2, and Section 9.1.2, 'Spent Fuel Storage,' Revision 3, July 1981.

9. Letter from J. H. Ferguson (Virginia Electric and Power Company) to H. R. Denton (U.S. Nuclear Regulatory Commission), "Amendment to Operating Licenses, Surry Power Station Unit Nos. 1 and 2, Proposed Technical Specification Change No. 85," Serial Number 400, May 15, 1980.

10. Letter from B. R. Sylvia (Virginia Electric and Power Company) to H. R. Denton (U.S. Nuclear Regulatory Commission), "Additional Information, Surry Power Station Unit Nos. 1 and 2, Proposed Technical Specification Change No. 85," Serial Number 570, July 3, 1980.

11. Letter from B. R. Sylvia (Virginia Electric and Power Company) to H. R. Denton (U.S. Nuclear Regulatory Commission), "Additional Information, Surry Power Station Units No. 1 and No. 2, Proposed Technical Specification Change No. 85," Serial Number 773, September 15, 1980.

12. Letter from B. R. Sylvia (Virginia Electric and Power Company) to H. R. Denton (U.S. Nuclear Regulatory Commission), "Additional Information, Surry Power Station Units No. 1 and No. 2, Proposed Technical Specification Change No. 85," Serial Number 773A, September 23, 1980.

13. Letter from B. R. Sylvia (Virginia Electric and Power Company) to H. R. Denton (U.S. Nuclear Regulatory Commission), "Additional Information, Surry Power Station Units No. 1 and No. 2, Proposed Technical Specifications Change," Serial Number 960, December 4, 1980.

14. Letter from B. R. Sylvia (Virginia Electric and Power Company) to H. R. Denton (U.S. Nuclear Regulatory Commission), "Supplementary Information for Surry Power Station Units No. 1 and 2 and North Anna Power Station Units No. 1 and 2 Proposed Technical Specifications Change," Serial Number 195, March 26, 1981.

15. Letter from R. H. Leasburg (Virginia Electric and Power Company) to H. R. Denton (U.S. Nuclear Regulatory Commission), "Supplementary Information, Surry Power Station Units No. 1 and No. 2 and North Anna Power Station Units No. 1 and No. 2," Serial Number 432, July 24, 1981.

16. Letter from W. L. Stewart (Virginia Electric and Power Company) to H. R. Denton (U.S. Nuclear Regulatory Commission), "Extended Burnup Operation to 45,000 MWD/MTU Batch Average Burnups," Serial Number 678, November 23, 1983.

17. Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Surry Power Station Units 1 and 2 - North Anna Power Station Units 1 and 2 - Request for Removal of 45,000 MWD/MTU Batch Average Burnup Restriction," Serial Number 92-743, November 25, 1982.

18. Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Request for Removal of 45,000 MWD/MTU Batch Average Burnup Restriction," Serial Number 92-743, November 25, 1992.

19. S. L. Davidson and D. L. Nuhfer (Eds.), "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 and Appendices A through D, June 1990.

20. Letter from W. R. Cartwright (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "North Anna Power Station Units No. 1 and 2 - Proposed Technical Specification Change - Fuel Enrichment Increase," Serial Number 88-603, September 30, 1988.

21. Oak Ridge National Laboratory, "KENO-V.a An Improved Monte Carlo Criticality Program with Supergrouping," ORNL-NUREG-CSD-2-VI-R2, December 1984.

22. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," ORNL-NUREG-CSD-2-VI-R3, December 1984.

23. Stone & Webster Engineering Corporation, "Arrg't Fuel Building- Sheet 1 - Surry Power Station," Drawing Number 11448-FM-9A-10.

24. Stone & Webster Engineering Corporation, "New Fuel Racks - Guide Tube Assemblies," Drawing Number 11448-FM-46A-2.

25. 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.

26. 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.

27. W. D. Newmyer, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-P, June, 1995.

28. Letter from T. E. Collins (U.S. Nuclear Regulatory Commission) to Tom Greene (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, 'Westinghouse Spent Fuel Rack Criticality Analysis Methodology'," TAC NO. M93254, October 25, 1996.

29. "American National Standard for Nuclear Criticality Safety Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983.

30. U.S. Nuclear Regulatory Commission, "Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross Section Library for Use in Criticality Safety Analyses", NUREG/CR-6102, ORNL/TM-12460, March, 1995.

31. Robert E. Odeh, "Tables for Normal Tolerance Limits, Sampling Plans, and Screening," Department of Mathematics, University of Victoria and D.B. Owen Department of Statistics, Southern Methodist University.

32. Technical Specifications - Surry Power Station, Units Number 1 and 2, Dockets 50-280 and 50-281, through Amendment No. 212, July 15, 1997.

33. Updated Final Safety Analysis Report - Surry Power Station, Units 1 and 2, Docket Nos. 50-280 and 50-281.