VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 27, 2000

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

Serial No.	00-491
NL&OS/GSS/ETS	R0
Docket Nos.	50-338
	50-339
License Nos.	NPF-4
	NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGES
INCREASED FUEL ENRICHMENT AND
SPENT FUEL POOL SOLUBLE BORON AND FUEL BURNUP CREDIT

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Virginia Power) requests amendments, in the form of changes to the Technical Specifications to Facility Operating Licenses Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes will 1) increase the fuel enrichment limit to 4.6 weight percent Uranium^{235}, 2) establish Technical Specifications Limiting Conditions for Operations for Spent Fuel Pool (SFP) boron concentration and fuel storage restrictions, and 3) eliminate the value for the allowance for uncertainties in the calculation for Keff in the SFP criticality calculation. A discussion of the proposed Technical Specifications changes, including the criticality and boron dilution analyses, is provided in Attachment 1. The mark-up and typed pages are provided in Attachments 2 and 3, respectively.

The criticality analysis of spent fuel racks was performed to permit a maximum North Anna fuel assembly enrichment of 4.6 weight percent Uranium ^{235}. The reanalysis was performed in accordance with "Westinghouse Spent Fuel Rack Criticality Analysis Method," WCAP 14416-NP-A, Revision 1, dated November 1996 with modifications. The methodology incorporates boron, burnup and geometry credits and eliminates the spent fuel rack Boraflex credit in the criticality analysis. During preparation of the package, Virginia Power became aware that the axial burnup shape reactivity treatment in the WCAP is potentially nonconservative. The axial burnup shape reactivity bias issue that was identified was specifically addressed in the attached criticality analysis for North Anna.

We have evaluated the significant hazards considerations and have determined that the proposed Technical Specifications changes do not constitute a significant hazards consideration as defined in 10 CFR 50.92. The basis for our determination that the

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changes do not involve a significant hazards consideration is provided in Attachment 4. We have also determined that operation with the proposed changes will not result in any significant increases in the amounts of effluents that may be released offsite and in any significant increases in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed changes.

Due to the lead time required to manufacture fuel and design the reload core at the new enrichment, it is requested that the NRC approve the proposed Technical Specification changes by May 1, 2001 to support a Fall 2001 refueling for Unit 1. In addition, due to the extensive fuel shuffle required to implement the fuel storage restriction, a ninety-day implementation period is requested.

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

Very truly yours,

Leslie N. Hartz
Vice President - Nuclear Engineering and Services

Attachments:

1. Discussion of Changes
2. Mark-up of Technical Specifications Changes
3. Proposed Technical Specifications Changes
4. Significant Hazards Consideration Determination

Commitments made in this letter:

1. None

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COMMONWEALTH OF VIRGINIA)
)
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The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering & Services, of Virginia Electric and Power Company. She has affirmed before me that she 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 her knowledge and belief.

Acknowledged before me this 27th day of September, 2000.

My Commission Expires: 3/31/04.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Discussion of Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

DISCUSSION OF CHANGES

1.0 INTRODUCTION

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Virginia Power) proposes to establish Technical Specifications requirements for spent fuel pool (SFP) soluble boron concentration and fuel assembly loading restriction based on burnup and enrichment. These new Technical Specifications requirements will permit elimination of the Boraflex credit from the spent fuel pool criticality calculations. In addition, we propose increasing the maximum fuel enrichment for North Anna Units 1 and 2 from the current Technical Specifications limit of 4.3 weight percent U^{235} to 4.6 weight percent U^{235}. This higher enrichment has been incorporated into the new criticality calculations for the spent fuel pool and fresh fuel storage racks.

The increase in enrichment, in conjunction with the previously submitted Technical Specifications change (Serial No. 00-305) to increase the soluble boron concentration in plant safety systems submitted June 22, 2000, (Reference 1) will reduce the need for extended periods of reduced power operation (coastdowns) at the end of each operating cycle and permit fuel discharge burnups more compatible with the current lead rod limit of 60,000 MWD/MTU. This will help optimize fuel cycle costs while continuing to satisfy the current core power distribution and safety limits.

2.0 BACKGROUND

Due to gamma irradiation and exposure to a wet pool environment, degradation of the Boraflex panels was identified at several utilities. In June, 1996, the NRC issued Generic Letter 96-04 (Reference 2) which requested that licensees with spent fuel storage racks containing Boraflex provide an assessment of the physical condition of the Boraflex and verify that a subcriticality margin of 5% can be maintained in the spent fuel pool. Blackness tests were conducted at the North Anna Power Station in January of 1997. These tests and subsequent analysis confirmed that the 5% subcriticality margin can be maintained at North Anna (Reference 3). However, recognizing that further degradation of the Boraflex will occur, we have committed to reinspecting the condition of the Boraflex at 5 year intervals to ensure that the design basis of the spent fuel pool is maintained. The elimination of the Boraflex credit in the spent fuel pool criticality calculations will eliminate this periodic blackness testing requirement before the Boraflex must be reinspected (January 2002).

Although no credit is taken for the presence of the Boraflex in the spent fuel racks in the criticality analyses, the Boraflex panels will remain in their current locations in the racks. The degree of solubilization of the Boraflex can be monitored by long term trending of spent fuel pool silica levels. The Virginia Power chemistry program provides for periodic testing of the North Anna spent fuel pool silica level because of the presence of the Boraflex in the SFP.

2.1 Design and licensing basis

2.1.1 Spent Fuel Pool and Storage racks

The storage racks in the spent fuel storage facility are comprised of double walled Type 304 stainless steel boxes welded to each other by tie plates. The double walled construction provides a compartment into which B_4C neutron absorber elements (Boraflex) have been added. These elements are positioned on each side of the storage cell at an elevation corresponding to the fuel region.

The SFP storage racks are currently designed to maintain K_{eff} to less than 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties and a nominal 10 9/16" center-to-center distance between fuel assemblies placed in the storage racks. The current capacity of the North Anna SFP storage racks is 1737 fuel assemblies with maximum enrichment of 4.3%. There are currently no storage location restrictions based on enrichment or burnup.

2.1.2 Fuel Assembly Enrichment

In September 1988, the Virginia Electric and Power Company requested an increase in the maximum fuel enrichment for the North Anna Units 1 and 2 from 4.1 to 4.3 weight percent U^{235} to allow average batch burnups to reach 45,000 MWD/MTU (Reference 4). The 4.3 weight percent U^{235} enrichment was approved by the NRC in June 1990 (Reference 5). In December 1993, the NRC approved operation to batch average burnups of 50,000 MWD/MTU or above, as long as the maximum rod average burnup does not exceed 60,000 MWD/MTU (References 6 and 7).

3.0 PROPOSED CHANGES

Currently, the maximum reload fuel enrichment permitted by North Anna's Technical Specifications is 4.3 weight percent U^{235}. Virginia Power proposes to increase the maximum reload fuel enrichment to 4.6 weight percent U^{235}.

In addition, Virginia Power proposes to change the method of criticality calculation used to evaluate the effect of a fuel enrichment change on its spent fuel pool. Specifically, we will eliminate credit for Boraflex in the spent fuel pool criticality calculations and will instead take credit for soluble boron and for fuel loading restrictions which are based on fuel enrichment and burnup. To ensure that the bases of this analysis are maintained, new Technical Specifications will be added to control the boron concentration in the spent fuel pool, and imposing restrictions on the storage locations for some fuel assemblies.

The proposed revisions to the maximum reload fuel enrichment and spent fuel pool and storage racks may be implemented by the following modifications to the North Anna Units 1 and 2 licenses and Technical Specifications:

3.1 Fuel Enrichment

1. Technical Specification 5.3.1 Reactor Core, Fuel Assemblies

   The specifications on the reactor core fuel assemblies will be changed to increase the maximum enrichment of the reload fuel from 4.3 to 4.6 weight percent U^{235}.

3.2 Spent Fuel Pool and Storage Racks

1. New Technical Specification 3.7.14 Fuel Storage Pool Boron Concentration

   The spent fuel pool boron concentration shall be maintained $\geq$ 2500 ppm. A surveillance requirement will be added to ensure that this concentration is measured on a weekly basis.

2. New Technical Specification 3.7.15 Spent Fuel Assembly Storage

Technical Specification 3.7.15 is being added to require that the combination of initial enrichment and burnup of each spent fuel assembly stored in the spent fuel pool be in accordance with the following restrictions:

- New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Acceptable" burnup domain of Figure 3.1 may be stored in the spent fuel pool in a non-matrix location (intermediate reactivity location) or a low reactivity location in the 5x5 matrix configuration shown in Figure 4.4. It may also be placed in a high reactivity location if stored in the 5x5 matrix configuration shown in Figure 4.4.

- New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Conditionally Acceptable" domain of Figure 3.1 may be stored in the spent fuel pool in a non-matrix location (intermediate reactivity location), but must be placed in a high reactivity location if stored in the 5x5 matrix configuration shown in Figure 4.4.

- New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Unacceptable" domain of Figure 3.1 must be stored in the spent fuel pool in a high reactivity location in the 5x5 matrix configuration shown in Figure 4.4. A fuel assembly transferred from Surry for storage in the North Anna spent fuel pool must be treated as a fuel assembly in the "Unacceptable" domain.

In addition, appropriate surveillance requirements and corrective actions are identified.

3. Technical Specification 5.6.1 Fuel Storage, Criticality

The specification for the spent fuel storage rack criticality will be modified to require that the K_{eff} of the spent fuel storage racks be less than 1.0 instead of 0.95 when flooded with unborated water. A specification is added that will require the K_{eff} of the spent fuel storage racks be less than 0.95 when flooded with borated water of 350 ppm, which includes allowances for uncertainties but not for postulated accidents.

4.0 TECHNICAL AND SAFETY EVALUATION

4.1 Core Physics

The core physics calculations are only affected by the proposed increase in maximum fuel enrichment. The fuel management strategy will continue to be a very low leakage modified checkerboard pattern with the fresh fuel preferentially loaded in the interior of the core, intermixed with once-burned fuel. The peripheral assemblies will typically continue to consist primarily of twice-burned fuel with limited use of some once burned fuel. For purposes of determining the properties of the burned fuel, it was assumed that the same burnable poison product currently in use will be utilized for the higher enrichment cores. A higher boron enrichment may be necessary in the burnable poison product to control peaking with higher enrichment fuel, especially during the transition cycles.

4.1.1 Core Parameters

Recent North Anna cycle load factors have approached 100% and refueling outages have been reduced in length, resulting in fuel cycles with excessively long end of cycle power coastdowns. The primary motivation for increasing feed enrichment is to reduce the length of end of cycle power coastdown. Scoping calculations indicate that an increase in enrichment to an average of 4.56 w/o will be sufficient to eliminate power coastdown under current operating assumptions. The impact on core physics parameters is summarized below.

The assumed equilibrium fuel loading for the scoping calculations is a split batch loading of 28 feed assemblies at 4.1 w/o U^{235} and 36 feed assemblies at 4.25 w/o U^{235} for current cycles, and 28 feed assemblies at 4.5 w/o U^{235} and 36 feed assemblies at 4.6 w/o U^{235} for proposed future cycles. Each cycle is 18 months long with a 35 day refueling outage and 98% load factor. No changes to the fuel loading strategy (very low leakage) or current burnable poison loadings were assumed.

The anticipated impacts of enrichment increase on core physics parameters are as follows:

Critical boron increase: $\leq$ 400 ppm increase (with no burnable poison changes)
Radial and axial peaking factors: No significant change for equilibrium cycles
Most positive moderator temperature coefficient (MTC): +2 pcm/°F more positive
Most negative MTC: -1 pcm/°F more negative
Doppler temperature coefficient: No significant change
Control rod worth beginning of cycle (BOC): -5% (less worth)
Control rod worth end of cycle (EOC): -2% (less worth)
Differential boron worth (BOC): -8% (less worth)

These changes are not expected to require any changes to existing limits, with the exception of the critical boron concentration increase. Cycle values for FdH, most positive MTC, and shutdown margin (SDM) are expected to approach the current limits either during transition to higher enrichments (FdH) or after reaching equilibrium (MTC, SDM). The critical boron concentration is evaluated as part of the reload design process, so the impact of any required increase in this parameter will be calculated prior to each reload to ensure compliance with all applicable Technical Specifications.

The impacts of the enrichment increase will be explicitly included in the predictions used for certification of surveillance limits, startup physics testing, and for the physics parameter operator curves used as input to station procedures. Based on the comparison of physics parameters for the higher enriched cores and the fact that all the differences will be explicitly modeled, no significant impact on surveillance limits, startup physics testing, and core operation is expected due to the enrichment increase.

4.1.2 Impact on Reactor Vessel Fluence

The fuel management strategy is expected to remain the same for the higher enriched cores. The current fuel loading strategy is a low leakage pattern with primarily twice-burned assemblies and a smaller number of once burned assemblies on the core periphery. With the use of higher enrichments, the size of reload batches may be decreased slightly and cycle coastdowns may be reduced, which in turn will result in marginally higher burnup fuel on the periphery. The reactivity of the peripheral fuel will therefore be about the same or lower than in current cores, so the influence on vessel fluence (which is primarily affected by the peripheral fuel assemblies) will be small.

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 peripheral 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.

4.1.3 Axial Offset Anomaly (AOA)

Industry data suggest that using higher boron concentrations (due to higher fuel enrichments) may increase the likelihood of developing an axial offset anomaly. In addition, coolant chemistry also plays a role in the development of AOA, and it is possible that some coolant chemistry changes could increase the probability of developing this undesirable phenomenon. However, some of the conditions present in plants that have experienced an AOA event will still not be present at North Anna.

Fresh to fresh assembly interfaces are a potential AOA promoter. However, the fuel loading strategy for the higher enrichment fuels at North Anna is similar to the current strategy which involves a checkerboard type pattern with minimal fresh to fresh assembly interfaces. Also, the current plan is not to increase the FdH and Fq limits at North Anna which means our peaking will remain well below the levels seen in plants with the AOA phenomena. Finally, the transition to higher enrichment fuel in the core will be a gradual transition where boron concentrations and fuel enrichments incrementally increase. If indications of AOA are observed, subsequent reloads will be modified by backing off of these parameters in following reloads.

4.1.4 Incore Core Power Distribution and Monitoring

The reactivity impact of the higher enrichments on the core design will be accounted for by explicitly modeling the changes in the core design models. The current power distribution limits will govern the core design development keeping radial and axial power peaking factors consistent with current core designs. Higher beginning of cycle boron concentrations and shorter coastdowns will tend to shift the BOC axial offsets in a more positive direction which can be controlled with a small amount D-bank insertion if necessary. The changes in the signals received by the incore detector system during flux map evaluations due to the higher enriched fuel are expected to be in line with typical changes observed between reloads.

4.1.5 Excore Power Monitoring

The source range detector (SR) count rate for core onload is not expected to change significantly with the use of higher fuel enrichments since it is primarily a function of assembly burnup. Assembly burnups are expected to slightly increase. This would tend to increase the measured count rate, which is beneficial. For the other uses of the SR along with the intermediate range detectors (IR) and power range detectors (PR), the primary influences on the excore detector responses are the assembly nearest the detectors and the control rod position. Since the fuel loading strategy will remain the same for the higher enrichment cores, the fuel assemblies nearest the excore detectors will be very similar in reactivity and influence to those in the current cores.

In addition, as noted in Section 4.1.1, the control rod worths will not change significantly (<5%). Predicted excore signals will explicitly account for the impact of higher fuel enrichments on detector signals. Overall, the impact of the higher enrichment cores on excore power monitoring is expected to be similar to the perturbations seen from reload to reload, and will be explicitly modeled in the predictions provided to North Anna.

4.2 New Fuel Storage Area

The requirements for the criticality analysis of new fuel storage are given in Section 9.1.1 of the standard review plan (NUREG-0800; Reference 8). Since no physical modifications are being made to the current North Anna Fresh Fuel Storage Racks (FFSR), the current analysis addresses only the impact of the increased enrichment on the requirement of subcriticality under normal and postulated abnormal rack conditions (General Design Criteria 62). According to NUREG-0800, Section 9.1.1, the design of the FFSR must be such that the K_{eff} will not equal or exceed 0.95 assuming the FFSR is dry or flooded with unborated water, and will not exceed 0.98 assuming optimum moderation.

4.2.1 Model Data

The FFSR was modeled in three dimensions to minimize unnecessary conservatism and uncertainty. The SCALE 4.3 code package and in particular, the KENO-V.a computer code was used for the criticality calculations. The ENDF/B-V 238-group cross sections were used for the KENO calculations. The accuracy of the cross sections using these codes has been documented. A simplified 3D KENO model was used for uncertainty cases (rack cell pitch, fuel enrichment and fuel density).

The new fuel storage area at North Anna is common to North Anna Units 1 and 2. The FFSR at North Anna consists of nine parallel rows of storage cells with a total of 126 storage locations. Figure 4.1 shows the top down view of the North Anna FFSR layout. Figure 4.2 shows the side view of an individual storage cell as modeled. Each storage location consists of a 9 inch square (inside measure) stainless steel box (storage tube), 165 inches tall with walls 1/8 inches thick. The storage area walls and floor are made of concrete. The storage area is normally dry and designed to drain in the event of inadvertent introduction of liquid into the area. North Anna FFSR design data is summarized in Table 4.1. The fuel assembly dimensions and material data used in the analysis are summarized in Table 4.2.

The neutronic properties of several fuel assembly parts and rack components have been neglected in this model for simplicity and conservatism. The fuel assembly top and bottom nozzles (SS-304), grids (Inconel, Zircaloy or ZIRLO), sleeves (SS-304, Zircaloy or ZIRLO) and all storage rack structural materials other than the storage tube were modeled as moderator regions. This is acceptable from a criticality standpoint because the omitted materials, particularly the steel and the Inconel components, are significant neutron absorbers. In addition, the ZIRLO fuel assembly was modeled as Zircaloy. This is also conservative because the use of Zircaloy produces a slightly higher calculated K_{eff}.

In addition, it was assumed that there was no U^{234} or U^{236} in the fuel. This was a conservative assumption because these uranium isotopes are neutron absorbers whose presence would reduce the calculated K_{eff}. The fuel assemblies were also modeled without any burnable poisons, control rods or other insert components that would reduce K_{eff}.

4.2.2 Conditions Modeled

Initially, the nominal FFSR configuration was modeled. In order to confirm that K_{eff} remains below 0.98 even during conditions of optimum moderation, the nominal configuration was flooded with moderator (water) of different densities. This provided the peak K_{eff} regions in the K_{eff} vs moderator density function. Geometric and fuel design tolerances were also evaluated at moderator densities equal to the density of the peak K_{eff}.

Because the FFSR is designed for dry storage, a fully flooded condition is one of two accident scenarios. The other scenario is the dropping of a fuel assembly which can result in the fuel being compacted in a storage cell. The flooding accident results in the largest increase in K_{eff}. Therefore, the handling accident was not analyzed in this calculation. The flooding accident was

conservatively incorporated into the K_{eff} vs moderator density function used for the calculation of the nominal K_{eff}, so no separate accident scenario calculation was performed.

4.2.2.1 Nominal Configuration

The base configuration for the analysis consisted of a fully loaded storage area of 126 fresh assemblies, with nominal density fuel enriched to 4.6 weight percent U^{235}, and nominally centered in the storage cans. The air region in the storage area was modeled as water vapor at a density of 0.0009983 g/cc.

4.2.2.2 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. In addition a 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 0.0009983 g/cc to 0.9983 g/cc.

4.2.2.3 Tolerance Cases

Reference 9 lists the tolerance and uncertainty cases to be considered for spent fuel storage rack analyses. It is prudent to also consider these for the FFSR as applicable. The following tolerances were analyzed for the North Anna FFSR:

1) Enrichment tolerance of ±0.05 w/o U^{235} about the nominal fresh reference enrichments.

2) Variation of ±2.0% about the reference UO_2 theoretical density.

3) Variation in the fuel pellet dishing fraction from 0% to twice the nominal dishing.

4) Tolerance about the nominal storage cell center-to-center pitch and material thickness.

5) Asymmetric positioning of fuel assemblies within the storage cells.

Of these, items 2 and 3 were combined into one case in the conservative direction. Items 4 and 5 were also combined by simply reducing the pitch of all storage locations by one inch. This tolerance was not applied randomly, but was cumulatively applied to all storage locations. This represents a maximum accumulated error of 15 inches across the long axis of the FFSR and 7

inches across the short axis. Because this tolerance was applied conservatively as an accumulated error, the pitch tolerance case is bounding for the combination of pitch and assembly placement tolerance.

4.2.3 Results of Criticality Calculations for the Fresh Fuel Storage Racks

The maximum (worst case) K_{eff} is calculated as follows:

$$K_{eff} = K_{nominal} + B_{method} + B_{uncertainty}$$

$K_{nominal}$ = K_{eff} using nominal design data at optimum moderation (peak K_{eff}),
B_{method} = method bias determined from benchmark critical calculations
$B_{uncertainty}$ = bias from the geometric and fuel design tolerances

4.2.3.1 Peak K_{eff}

K_{eff} as a function of moderator density reaches a peak at a water density of 0.0649 g/cc. This is the optimum moderation case. Figure 4.3 shows the K_{eff} vs water density curve while Table 4.3 summarizes each case. Note that the water density is shown in terms of water volume fraction. A water volume fraction of 1.0 corresponds to a water density of 0.9983 g/cc and a water volume fraction of 0.0 corresponds to a water density of 0.0 g/cc. Note also that there are two distinct peak K_{eff} regions, at approximately 0.065 g/cc and 0.9983 g/cc (fully flooded).

4.2.3.2 Effect of calculational uncertainty and bias

Table 4.4 shows the results from the tolerance cases. In this table, the reactivity difference, dK, is calculated for the reference case as 1.65 * $S.D._{Reference}$, while dK is calculated for the tolerance cases using the following equation:

$$dK = K_2 - K_1 + 1.65 * \sqrt{(S.D._1)^2 + (S.D._2)^2}$$

Where K_1 = K_{eff} of the reference case,
K_2 = K_{eff} of the tolerance case,
$S.D._1$ = Standard Deviation of the reference case, and
$S.D._2$ = Standard deviation of the tolerance case.

The total uncertainty is determined by taking the root mean square of the individual tolerance and methodology uncertainty dKs. The method bias is then added to the total uncertainty. Along with the tolerance case results, Table 4.4 shows the calculation of the total uncertainty and bias for the FFSR.

4.2.3.3 Worst Case Keff

Using the peak K_{eff} and tolerance case results, the worst case K_{eff} is calculated as follows:

Peak K_{eff}:	0.90726
Method Bias:	0.00357
Total Uncertainty:	0.02515
Maximum K_{eff} (Worst Case):	0.93598

This maximum K_{eff} includes conservative allowances for bias and uncertainty, and is bounding for normal dry storage and for the postulated complete flooding with water moderator. Therefore, the K_{eff} limits of 0.95 for dry conditions and 0.98 for optimum moderation will both be satisfied for storage for fuel enriched to 4.6 w/o U^{235}.

4.3 Spent Fuel Storage

Criticality of fuel assemblies in the spent fuel storage racks (SFSR) is prevented by the design of the rack and by administrative controls related to boron and burnup credit, which limits fuel assembly interaction. The North Anna spent fuel racks have been reanalyzed to allow for an increase in the maximum enrichment to 4.6 weight percent U^{235}. This reanalysis also does not take credit for the presence of Boraflex. The Westinghouse Owners Group (WOG) approach (Reference 9) incorporating boron credit, burnup credit and geometry credit was used, with some modifications. These modifications to the WOG approach are discussed in Section 4.3.2.

4.3.1 Spent Fuel Pool Design Description

The spent fuel storage pool at North Anna consists of three parallel rows of storage racks with a total of 16 racks, for a capacity of 1737 fuel assemblies. The rack storage cell configurations are 9x9 (3 racks), 9x12 (1 rack), 10x11 (5 racks), 10x12 (6 racks) and 11x11 (1 rack). Five potential cell locations in one rack were allocated to accommodate storage of 2 failed fuel canisters, reducing the total potential storage locations in the SFSR from 1742 to the licensed capacity of 1737 assemblies.

4.3.2 Analytical Methods for Evaluation of Spent Fuel Storage Racks

The SCALE 4.3 code package and in particular, the KENO-V.a code was used for most of the SFSR calculations. ENDF/B-V 238-group cross sections were used for the calculations with the exception of the axial burnup effect determination. Axial burnup effect calculations were performed using the 44-group ENDF/B-V cross sections to reduce computer system requirements. A pair of ENDF/B-V 238-group cases were run to verify the adequacy of the 44-group cross sections for that purpose.

The Westinghouse PHOENIX code was used for certain sensitivity cases (temperature bias) and for determining the burnup credit curves. References relevant to use and accuracy of the PHOENIX code are contained in Reference 9.

The fuel in the SFSR was considered to be axially infinite. The effect of axial burnup distributions (3D) versus the uniform average burnup (2D) was calculated and accounted for in the burnup credit requirements.

Two different cell configurations were used in the analysis. The first was a 5x5 cell cluster in which one cell with no fuel is surrounded by up to four "high" reactivity assemblies (4.6 w/o U^{235}). "Low" reactivity fuel (1.56 w/o U^{235}) fills the other 20 locations. This configuration is referred to as the 5x5 matrix. The second configuration was a 3x3 non-matrixed cell cluster. All 9 cells in the 3x3 cluster are assumed filled with fuel of "moderate" reactivity (2.0 w/o U^{235}). Figure 4.4 is a representation of the configurations modeled in the criticality analysis. The pool model is effectively infinite in the x-y plane as well as axially because both the 5x5 storage cell model and the 3x3 model used periodic x-y boundary conditions. For the matrix configuration, this means that the 5x5 cell clusters cannot overlap: i.e., the outer row of the low reactivity fuel may not serve as the outer row of two adjacent matrix clusters. It is acceptable for the 5x5 and the 3x3 configurations to adjoin one another.

At the pool boundaries, it is acceptable to place a partial 5x5 matrix configuration. The leakage out of the boundary will make this partial 5x5 matrix bounded by the nominal 5x5 matrix configuration.

The design method used to insure that a subcritical condition was maintained in the SFSR was similar to the approved Westinghouse Owners Group (WOG) methodology, which is described in Reference 9. The primary differences between the method used and the WOG methodology were the following:

1) A reference theoretical density of 95.5% was used rather than the typical nominal value of 95%. The manufacturing tolerance associated with the theoretical density is maintained at a typical value of 2%.

2) The upper limit of nominal pool water temperature for the criticality analysis was 170°F rather than the value of 150°F typically used in Westinghouse analyses. This is based on UFSAR Section 9.13 requirements.

3) The static pressure of the pool water at the axial midpoint of the fuel is approximately 28 psia (due to the pressure of the water column over the fuel) rather than 14.7 psia. This results in a maximum temperature for the over-temperature accident of 246.41°F rather than the typical 212°F.

4) No bias for B^{10} self-shielding was required because the Boraflex panels are assumed to be replaced by water.

5) PHOENIX-P depletion calculations were not performed at the typical constant soluble boron concentration of 1500 ppm. A more conservative value was determined to account for the effects of power history, boron letdown and BP loading.

6) The geometric, fuel density and fuel enrichment sensitivity cases were all run using KENO. This is acceptable because KENO is adequately benchmarked for various fuel/water ratios, fuel enrichments and lattice dimensions. However, KENO is not benchmarked for temperature sensitivity or burnup, so the temperature bias and burnup credit data were calculated with PHOENIX.

American Society of Mechanical Engineering (ASME) equivalent steam tables were used to determine water density. The highest reactivity fuel was assumed to be 4.6 w/o U^{235} (Westinghouse 17x17 design). For simplicity and conservatism, no grids, nozzles or other structural materials in the fuel assemblies were modeled.

The WOG approach makes use of a dual criticality limit. The first limit is $K_{eff} < 1.0$ with no soluble boron. The second limit is $K_{eff} < 0.95$ with credit for soluble boron. The soluble boron is used to meet the K_{eff} limit and to mitigate the effect of abnormal configurations or accident conditions other than the dilution accident.

The first step in the WOG methodology is to calculate a SFP K_{eff} assuming no soluble boron but including allowances for the biases and uncertainties (including geometric and manufacturing tolerances). The nominal calculation is performed with KENO-V.a assuming fresh fuel (non-matrix or in a high/low reactivity matrix as appropriate). Use of a fuel enrichment higher than the maximum modeled fresh fuel enrichment will require credit for the fuel burnup, so the low enrichment fresh fuel actually represents burned fuel of equivalent reactivity. Burnup credit is calculated using the PHOENIX code (plus an allowance for axial burnup effects). K_{eff} is calculated using the following equation:

$$K_{eff} = K_{nominal} + B_{temp} + B_{method} + B_{self} + B_{uncertainty}$$

$K_{nominal}$ = K_{eff} at nominal conditions
B_{method} = method bias determined from benchmark critical comparisons
B_{temp} = temperature bias for normal operating range
B_{self} = B^{10} self shielding bias (Set to zero for NAPS with no Boraflex)
$B_{uncertainty}$ = tolerance bias

The second step of the WOG methodology is to perform the above calculation with sufficient soluble boron to ensure $K_{eff} < 0.95$. The tolerance and uncertainty calculations are also performed again with soluble boron.

The final step in the WOG methodology is to determine the amount of additional soluble boron credit (SBC) required to offset the maximum reactivity effects of postulated accidents and reactivity equivalencing uncertainty as outlined by the following equation:

$$SBC_{Total} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

$SBC_{95/95}$=soluble boron (ppm) required for $K_{eff} < 0.95$ on a 95/95 basis
SBC_{RE}=soluble boron (ppm) required for reactivity equivalencing methodology uncertainty
SBC_{PA}=soluble boron (ppm) required for $K_{eff} < 0.95$ under accident conditions.

Soluble boron is used to offset uncertainty associated with reactivity equivalencing (burnup credit). This includes the PHOENIX uncertainty of 0.01 ΔK per 30,000 MWD/MTU and an allowance for measured burnup uncertainty. The reactivity bias associated with the axial burnup distribution (3D calculation versus 2D axially infinite calculation) was treated as a bias in the calculation of the burnup requirement.

4.3.3 Criticality Analysis for Spent Fuel Storage Racks

This section describes the application of the analytical techniques and models employed to perform the criticality analysis for the North Anna spent fuel storage racks. It closely follows the WOG methodology steps outlined in Section 4.3.2. Section 4.3.3.1 describes the reactivity calculations performed to ensure a $K_{eff}<1.0$ assuming no soluble boron in the SFP. These calculations include allowances for the biases and uncertainties (including geometric and manufacturing tolerances). Section 4.3.3.2 describes the calculations to determine a sufficient soluble boron concentration to ensure $K_{eff} < 0.95$. The tolerance and uncertainty calculations were repeated with the applicable soluble boron concentration. Finally, Section 4.3.3.3 describes the determination of the additional soluble boron credit (SBC) required to offset the maximum reactivity effects of postulated accidents and reactivity equivalencing uncertainty.

4.3.3.1 No Soluble Boron Reactivity Calculations

A KENO-V.a model was developed for storage of fuel assemblies in the North Anna SFSR assuming no soluble boron (0 ppm). This model was based on the following assumptions:

1) 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 and Zircaloy or ZIRLO), sleeves (SS-304, Zircaloy or ZIRLO), the Boraflex absorber material, and all storage rack structural materials other than the storage can, the tie plates and the Boraflex wrappers are modeled as moderator or void (above the water level). This is acceptable because the omitted materials are either significant neutron absorbers (Boraflex, SS-304 nozzles and sleeves, Inconel grids) or have an insignificant effect on reactivity by design (Zircaloy or ZIRLO grids and sleeves).

2) For unirradiated fuel, no U^{234} or U^{236} is included. This is a conservative assumption because these uranium isotopes are neutron absorbers whose presence would reduce the calculated K_{eff}. This is consistent with the methodology in Reference 9.

3) The fuel assemblies have no burnable poisons, control rods or other insert components which would reduce K_{eff}.

4) ZIRLO fuel assemblies were modeled as Zircaloy. This is conservative because the use of Zircaloy produces a slightly higher calculated K_{eff}.

5) The methodology bias was calculated by determining the uncertainty about the KENO bias, instead of the uncertainty about the KENO K_{eff}. This approach is different from past North Anna spent fuel pool analyses, but is consistent with the WOG methodology (Reference 9).

6) Burnup credit cases assume no xenon, which is appropriate since xenon decays in approximately 4 days.

7) An infinitely reflected KENO model was used to represent the entire spent fuel pool. This modeling is conservative based on comparisons of infinitely reflected and actual pool model KENO K_{eff} for North Anna.

8) Physical dimensions and materials of the fuel assemblies and the spent fuel storage racks were modeled using nominal design data as given in Table 4.2.

9) The static pressure of the pool water was set to 28 psia, which is the approximate pressure at the middle of the fuel stack in the pool assuming approximately 20 feet of water above the fuel. Greater than 23 feet of water is required over the fuel per Technical Specification 3.9.11.

The KENO calculations were performed for a non-matrixed 3x3 cluster of cells and the 5x5 matrix configuration. In order to evaluate the reactivity effects of possible variations in geometry and material characteristics, tolerance calculations were performed with KENO. These tolerance cases included the following:

1) Enrichment tolerance of ±0.05 weight w/o U^{235} about the nominal fresh enrichment limits.

2) Variation of ±2.0% about the reference UO_2 theoretical density.

3) Variation in fuel pellet dishing fraction from 0% to twice the nominal dishing.

4) Tolerance about the nominal storage cell center-to-center pitch and material thickness.

5) Asymmetric positioning of fuel assemblies within the storage cells.

The rack storage cell pitch tolerance was determined in two different ways. The first reduced the storage cell pitch by 1/64 inch (1/8 inch over a 9x9 rack). The second reduced the storage cell pitch for one row by 1/8 inch in both the x and y dimensions.

KENO calculations were performed using both the 3x3 non-matrixed cell cluster and the 5x5 matrixed group of cells. For the non-matrixed configuration, the maximum best estimate K_{eff} accounting for all the uncertainties and bias is 0.99454 assuming fuel of reactivity equivalent to 2.0 w/o U^{235} new fuel. This leaves a margin of 0.00546 to the K_{eff} limit of 1.0 for an unborated pool. The no boron case for the non-matrixed fuel is summarized in Table 4.5.

The maximum best estimate K_{eff} for the 5x5 matrix configuration including uncertainty and bias was determined to be 0.99923 assuming a middle cell with no fuel, 4 cells surrounding the middle filled with fuel of reactivity equivalent to 4.6 w/o U^{235}, and the remaining cells in the matrix filled with fuel of reactivity equivalent to 1.56 w/o U^{235} new fuel. This gives a margin of 0.00077 to the K_{eff} limit of 1.0 for an unborated pool. This case is summarized in Table 4.6.

4.3.3.2 Soluble Boron Reactivity Calculations

A KENO-V.a model was developed for storage of fuel assemblies in the North Anna spent fuel storage racks assuming the presence of soluble boron. With this exception, this model was based on the same assumptions as for the zero boron model. Tolerance cases were explored for the soluble boron case in the same manner as for the zero boron case.

For normal conditions, a concentration of 230 ppm soluble boron was determined to be sufficient to keep the spent fuel pool K_{eff} below 0.95 when all uncertainties and tolerances are included. Table 4.7 summarizes the nominal and tolerance case results for the 5x5 matrix reactivity calculations. The boron dilution of the spent fuel pool was also analyzed and is discussed in Section 4.3.6.

For the 5x5 matrix cell, the maximum best estimate K_{eff} is 0.94908, giving a margin of 0.00092 to the borated pool K_{eff} limit of 0.95.

4.3.3.3 Calculation of Soluble Boron Concentration Requirements

Consistent with the approved WOG methodology, the amount of additional boron required to offset reactivity effects of accidents and reactivity equivalencing uncertainty was also determined.

4.3.3.3.1 Postulated Accidents

Two postulated accidents have been identified that affect the criticality calculations for the SFSR. One is the fuel misplacement accident and the other is the pool temperature increase accident. The fuel misplacement accident is by far the more limiting because it produces a bounding change in K_{eff}. Therefore, the temperature increase accident was not explicitly analyzed as part of the current evaluation.

Of all the possible single fuel misplacement scenarios in either the 3x3 or the 5x5 matrix, the most limiting fuel misplacement event is a fresh assembly of the maximum allowable enrichment placed in the center cell of the 5x5 matrix. Not only does this replace the lowest reactivity region in the matrix with the highest reactivity region but it also clusters the highest reactivity assemblies together. Table 4.8 presents the results of this fuel misplacement calculation. For the single misplaced assembly, a boron concentration of 780 ppm is required to maintain the SFSR K_{eff} below 0.95 including uncertainties and tolerances.

4.3.3.3.2 Reactivity Equivalencing Uncertainty

Burnup equivalencing is the process of determining a set of enrichment-burnup pairs which have the same $K_{infinity}$ in a rack geometry. The KENO reactivity calculations described above assumed fresh fuel of either 4.6, 2.0 or 1.56 w/o enrichment. The burnup credit calculations are necessary to determine what burnup must be reached for different initial enrichment fuel in order to reach the same reactivity as one of the three above enrichments.

The conditions experienced by the fuel influence the amount of Pu production, which has a direct impact on rack $K_{infinity}$. Some of the factors which harden the neutron spectrum and increase Pu production are higher soluble boron concentration in the reactor coolant system, higher discrete burnable poison loading and higher depletion power. A conservative boron concentration was therefore developed for the fuel depletion calculation considering limiting burnable poison loadings, radial power distributions and expected boron letdown.

The WOG methodology (Reference 9) indicates that 2D SFP calculations (axially infinite, average assembly burnup) are conservative up to about 40,000 MWD/T assembly burnup. However, other sources (References 10, 11 and 12) indicate this may not be entirely accurate. The combination of low relative burnup at the ends of the fuel and higher Pu production in the top of the fuel (due to higher moderator temperature) results in a reactive region toward the top of the fuel assembly. The redistribution of the flux to the higher reactivity region of the fuel and the resulting increase in axial leakage is the axial burnup effect.

To determine the impact of the axial effects, the 2D KENO $K_{infinity}$ was compared with the 3D 44 group KENO K_{eff} for both the non-matrix and the 5x5 matrix configurations. Figure 4.5 shows the graph of axial burnup reactivity bias as a function of fuel burnup for the non-matrix configuration. The reactivity difference between the 2D $K_{infinity}$ KENO and 3D K_{eff} KENO solution for 4 weight percent U^{235} fuel in the non-matrix configuration is shown. Similar reactivity differences provided in WCAP-14416 for 3 different weight percent U^{235} initial enrichment fuels (3 w/o, 4 w/o and 5 w/o) in the non-matrix configuration are also presented. This shows that above a burnup of approximately 12,500 MWD/MTU there is a larger non-conservative end effect for non-matrixed fuel than is indicated by WCAP-14416.

For the 5x5 matrix configuration, the reactivity difference between the 2D $K_{infinity}$ KENO and the 3D K_{eff} KENO solutions were calculated based on measured burnup distributions for North Anna fuel. The operation of this fuel was modeled through 3 operating cycles, and the reactivity difference was determined at multiple burnups, through about 49,000 MWD/MTU. From these calculations, there is no evidence of non-conservative end effects. The 2D $K_{infinity}$ at higher

burnups is greater than the corresponding 3D K_{eff}. One case was also re-run with ENDF/B-V 238 cross sections to verify the accuracy of the 44 group cross sections. There is a K_{eff} difference of about -0.008 dK (the 238 group K-effective is less than the 44 group K-effective), but the end effect is nearly the same.

Based on the KENO results, an end effect fit (conservatively set to zero if negative) was used to account for end effects for non-matrixed regions of burned fuel. The end effect was conservatively set to zero for the 5x5 matrix cases. Figure 4.6 summarizes the results of the PHOENIX cases for the 1.56 w/o reactivity equivalencing. No allowance for end effects is included because the 1.56 w/o fuel is located in the 5x5 matrix configuration. Figure 4.6 also contains the same information but for the 2.0 w/o fuel. For the 2.0 w/o fuel the end effects are included since this enrichment is used for determining fuel that will not be matrixed.

Fuel initially enriched to 4.6 w/o U^{235} must operate to a burnup of 43,669 MWD/MTU to have the same reactivity as fuel initially enriched to 1.56 w/o U^{235}. At this burnup, the PHOENIX uncertainty is 0.0146 ΔK. In addition, the measurement uncertainty associated with fuel burnup is 1965 MWD/MTU. The reactivity associated with this amount of burnup is 0.0109 ΔK. Therefore, the maximum burnup equivalencing uncertainty reactivity is 0.0255 ΔK.

A comparison of the 0 ppm and the 230 ppm soluble boron KENO base cases for the 5x5 matrix shows that 230 ppm soluble boron provides a reactivity reduction of 0.051 ΔK. From this relationship, the additional soluble boron credit required to offset burnup equivalencing uncertainty of 0.0255 ΔK is 115 ppm. This is rounded up to 120 ppm for conservatism.

4.3.3.3.3 Total Soluble Boron Requirements

The total soluble boron credit to ensure $K_{eff} < 0.95$ is 900 ppm. The total credit is composed of 230 ppm for normal conditions, an additional 550 ppm for accident conditions (single worst misposition in a 5x5 matrix) and 120 ppm for burnup equivalencing uncertainty.

4.3.4 Surry Spent Fuel

The shipment of Surry spent fuel assemblies of up to 4.1 weight percent U^{235} for storage in the North Anna spent fuel pool is permissible under the current North Anna operating license. The maximum permissible enrichment of the shipped Surry spent fuel assemblies is 0.5 weight percent less than the proposed North Anna maximum reload fuel enrichment. Any such transfer would be for burned fuel, which could not be stored at the Surry ISFSI. When determining the relative

reactivity of the Surry spent fuel, fresh Westinghouse 15x15 assemblies of 4.1 weight percent U^{235} are assumed for maximum possible reactivity.

A KENO single SFP storage location case shows that the K_{eff} reduction associated with the 0.5 weight percent far outweighs the small K_{eff} increase associated with the 15x15 design relative to the 17x17 design. The 15x15 design fuel assemblies of 4.1 weight percent U^{235} are less reactive than the 17x17 design fuel assemblies of 4.6 weight percent U^{235}. Therefore, it is acceptable to store Surry spent fuel assemblies in the high reactivity storage locations of the 5x5 matrix.

4.3.5 Additional Fueled Components

Fuel rod storage baskets are designed to hold up to 52 fuel rods and may be placed in spent fuel pool storage locations. The K_{eff} of a fully loaded storage basket inside a SFP storage location is compared to a low enrichment 17x17 fuel assembly stored in the same geometry. For the fuel rod storage basket, a full load of 52 fresh 4.6 weight percent 17x17 type fuel pins is assumed for maximum possible reactivity.

The K_{eff} of a fully loaded fuel rod storage basket is much lower than a 1.56 weight percent U^{235} fuel assembly. Therefore, the fuel rod storage basket may be stored in any matrix or non-matrix location where fuel is permitted. It may also be concluded that fewer than 52 fuel rods in a regular lattice will be less reactive than the 52 rod case, allowing for a partially loaded fuel rod storage basket to also be stored in any matrix or non-matrix location where fuel is permitted.

Loose fuel pellets and fuel pin fragments from failed fuel pins can be stored in containers in the North Anna SFP. Analysis shows that up to 2 fuel rods of loose fuel pellets/pin fragments (two fuel pins contain at least 540 pellets depending on fuel design) in a SFP storage location will produce a K_{eff} less than a K_{eff} of a spent fuel assembly with an enrichment of 1.56 w/o equivalent U^{235}. It can be concluded that 2 fuel rods of loose fuel pellets/pin fragments (less than or equal to 540 pellets) may be stored in any matrix or non-matrix location where fuel is permitted. The number of loose pellets and pellets in pin fragments stored in this manner is currently significantly less than 540.

Incore detectors are also stored in the North Anna spent fuel pool. These detectors are fission chamber detectors that contain a small amount of fissile material. Analysis confirms that up to 300 grams of highly enriched U^{235} can be stored in a spent fuel pool storage location and remain bounded by a K_{eff} of a 1.56 w/o U^{235} equivalent fuel assembly. Therefore, up to 300 grams of 100% enriched U^{235} (which exceeds the amount of U^{235} found in a fission chamber detector by orders of magnitude) may be stored in any matrix or non-matrix location where fuel is permitted.

In the spent fuel pool, there are two locations designed for failed fuel canisters. The spacing around fuel assemblies placed in these two locations is much greater than in all other racks in the North Anna spent fuel pool. Therefore, the analysis completed in section 4.3.3 bounds the failed fuel canister storage locations. For conservatism, the fuel stored in these locations is limited to fuel of 2.0 w/o U^{235} equivalent or less. This corresponds to a non-matrix region.

4.3.6 Decay Heat Load for Full Core Offload

An increase in the maximum fuel enrichment from 4.3 to 4.6 weight percent U^{235} by itself will not affect the decay heat load in the spent fuel pool during a full core offload. The determination of the decay heat load is primarily a function of the operational power and burnup and is not affected by the initial fuel enrichment. North Anna Units 1 and 2 will continue to be limited to a lead rod burnup of 60,000 MWD/MTU and thus the increased fuel enrichment will not affect the decay heat load currently used to evaluate the consequences of full core offload into the spent fuel pool.

Immediately after the core offload, the decay heat from the short-lived isotopes in the recently discharged fuel represents the major contribution to the spent fuel pool decay heat load. The production and decay of these short-lived isotopes tend to reach an equilibrium condition in the fuel during normal operation, so the concentration of these isotopes do not change significantly with increasing fuel burnup.

After a sufficiently long decay period, the longer-lived isotopes will provide the major source of decay heat within a fuel assembly. These isotope concentrations do increase with increased burnup, slightly increasing the fuel assemblies decay heat power. However, the contribution of such older fuel to the total spent fuel pool decay heat load is small, particularly for the limiting case discussed in the UFSAR.

In addition, the decay heat load on the spent fuel pool is checked as part of each reload safety evaluation. Therefore, if the use of a higher fuel enrichment indirectly leads to conditions that could adversely affect the decay heat load on the spent fuel pool, the change will be detected as part of the reload evaluation and a reassessment or redesign can be made.

A reassessment of the decay heat load will also be performed 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.

4.3.7 Misloading and Storage Configuration Control

Figure 4.4 shows the 5x5 matrix postulated to be used at the North Anna spent fuel pool. In this 5x5 scheme, the higher reactivity fuel (4.6 weight percent) can only be placed in certain locations. The other locations must be empty of fuel or filled with low reactivity fuel (1.56 weight percent U^{235} or equivalent) and the middle location must be empty of fuel. A fuel assembly misloading accident involves violation of these prescriptions. Non-fueled components or materials may be placed in the middle of the 5x5 matrix provided that analyses are performed to demonstrate that there is no associated increase in reactivity. To prevent misloading, tight controls will be maintained on fuel assembly positioning.

Fuel assembly positioning in the spent fuel pool is administratively controlled. Software to evaluate moves for assemblies into, from or within the pool is also under development to augment these administrative controls.

In addition, the spent fuel pool will be maintained with a concentration of soluble boron sufficient to maintain $K_{eff} < 0.95$ during a misloading accident. Under the proposed Technical Specifications changes the boron concentration in the spent fuel pool will be maintained at a minimum of 2500 ppm, which is much greater than the 900 ppm total soluble boron concentration that was determined in Section 4.3.3.3 to be sufficient to counter the misload accident scenario including uncertainties.

4.3.8 Spent Fuel Pool Dilution

The spent fuel pool criticality calculation determined that 900 ppm soluble boron will keep $K_{eff} <$ 0.95, including accident and burnup equivalencing scenarios. A Technical Specification is being added to require that the spent fuel pool boron concentration remain at or above 2500 ppm at all times. Thus, 2500 ppm is assumed to be the initial concentration for the spent fuel pool dilution event while 900 ppm is assumed to be the endpoint concentration.

Virginia Power has conservatively evaluated the potential for diluting the North Anna spent fuel pool. This evaluation indicates that a large volume of water is necessary to dilute the spent fuel pool from an initial nominal boron concentration of 2300 ppm to an assumed concentration of 1200 ppm. Although sources of water are available which could supply the required dilution volume, it would require hours of dilution flow into the spent fuel pool to achieve this degree of dilution. An even longer dilution time would be required to dilute the pool from the proposed Technical Specifications limit of 2500 ppm to below 900 ppm. With weekly surveillance of the

boron concentration in the pool, a level alarm on the pool and routine operator rounds through the fuel building where the SFP is located, it is considered non-credible that such a dilution flow could remain undetected by plant personnel long enough for sufficient dilution to occur that the criticality limits would be approached.

Additional conservatism is provided by proposed Technical Specification 3.9.1 (submitted June 22, 2000), which requires the boron concentration in the SFP to be greater than 2600 ppm during Mode 6 operation. It is likely that the spent fuel pool will be maintained at this level during other modes of operation. In practice, any dilution of the spent fuel pool would therefore take even longer than the scenario already evaluated. It is concluded that the dilution of the spent fuel pool is highly unlikely without being detected by plant personnel (See Appendix A for analysis of spent fuel pool dilution event).

4.4 Effects on Safety Analyses

4.4.1 Safety Analysis Inputs

Use of higher initial fuel enrichments would be expected to slightly increase the fuel temperatures and rod internal pressures used as input to accident analyses and other safety calculations. However, the fuel temperatures and fuel rod internal pressures used in the North Anna safety and accident analyses are generic values that are valid for fuel enrichments up to 5 weight percent U^{235} (Reference 13). The enrichment increase does not affect key core physics parameters that are important for safety and accident analyses. The change in the design basis for the spent fuel pool (elimination of credit for Boraflex and assuming credit for soluble boron) does not affect the inputs or assumptions of the safety analyses described in the North Anna UFSAR.

4.4.2 Radiological Consequences of Accidents

Evaluation of the radiological consequences of accident scenarios described in the North Anna UFSAR will not be affected by an increase in fuel enrichment or the changes in the design bases for the SFP. Assessments of the impact of increasing fuel burnup from 33,000 MWD/MTU to 60,000 MWD/MTU (References 14 and 15) by the NRC and fuel vendors demonstrated that radionuclide concentrations in the fuel are largely a function of fuel burnup rather than fuel enrichment. The current burnup limits on the North Anna units (Reference 7) will not be changing as a result of the 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 concentration of these isotopes in the fuel-clad gap is independent of the initial fuel enrichment. The isotopic concentration is primarily a function of the number of fissions that have occurred in the fuel, which is expressed as burnup, rather than the number of fissile atoms initially present in the fuel, which is expressed as fuel enrichment.

Increasing the fuel burnup has a limited effect on the short-lived isotope inventory in the fuel due to the development of an equilibrium condition between production and decay. Instead, extended burnups increase the fraction of the short-lived isotopes that migrate into the fuel-clad gap region (Reference 15). North Anna Units 1 and 2 will continue to be limited to a lead rod burnup of

60,000 MWD/MTU and thus the increased fuel enrichment will not affect the source terms currently used to evaluate the consequences of postulated accident conditions.

The analysis of the source term inventory for extended burnup fuels has been conducted with assemblies with fuel cladding made of Zircaloy. However, North Anna fuel assemblies are currently constructed with cladding and structural components made of the ZIRLO alloy. ZIRLO replaced Zircaloy for the purpose of enhancing the reliability of the clad and extending fuel burnup levels.

Westinghouse has addressed the performance of the ZIRLO alloy in fuel assemblies in Reference 16. This assessment concludes that the core inventories from fuel cladding of Zircaloy or ZIRLO are comparable. There is no change in the core inventory due to a change to the ZIRLO alloy. Any analysis completed on fuel with Zircaloy is therefore applicable to ZIRLO clad fuel assemblies.

The source term for a storage cask handling accident is different from the above discussion. This is due to the decay of the radioactive isotopes that has occurred during the residence time of the fuel assemblies in the spent fuel pool.

After 150 days in the spent fuel pool, the activity of the shorter-lived isotopes in the fuel decays to very low levels. The longer-lived isotopes, such as Kr^{85}, take over as the dominant contributors to the accident source term. As with the shorter-lived isotopes, the concentration of the longer-lived isotopes in the fuel-clad gap region is not dependent on the initial fuel enrichment but rather is determined by the fuel burnup.

Extended burnups increase the total concentration of these longer-lived isotopes in the fuel as well as increasing the fraction of such isotopes that migrate into the gap region, thereby increasing the source term. However, North Anna Units 1 and 2 will continue to be limited to a lead rod burnup of 60,000 MWD/MTU and thus the increased fuel enrichment will not affect the source terms currently used to evaluate the consequences of postulated cask handling accident conditions.

4.4.3 Safety Analysis Limits

Other than the proposed changes to those Technical Specifications which deal with the maximum fuel enrichment and the addition of Technical Specifications on SFP soluble boron and SFP positioning, 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 (Reference 17). 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 the amount of U^{235} initially present in the fuel. No change in the lead rod burnup limit of the North Anna fuel is requested with this Technical Specifications change request, and any resolution of outstanding questions in this area will be unaffected by the increase in the initial enrichment to 4.6 weight percent U^{235}.

4.5 Environmental Effects

The environmental effects of higher enrichment fuel and SFP criticality calculation changes are analyzed in terms of the radiological impacts, concerning normal and accident conditions, of various processes and activities. Radiological impacts are only indirectly affected by increasing fuel enrichment. The radiological impacts are primarily a function of operating power and burnup. The purpose of increasing the fuel enrichment is to produce the same power level for a longer period of time before refueling. Therefore, increased fuel enrichments will result in the same power level but increased burnups.

The environmental impact of operation of the North Anna units with high burnup fuel was previously addressed in References 4 through 7. Westinghouse has generically addressed the impact of extended burnup on the design and operation of Westinghouse fuel (Reference 14). In addition, the NRC had an independent assessment conducted (Reference 15) of the environmental and economic impacts of the use of extended burnup fuels in light water power reactors.

The overall findings of these assessments were that no significant adverse effects would be generated by increasing the batch average burnup 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. These evaluations concerning the impact of extended burnup fuel are valid for an enrichment of 4.6 weight percent U^{235}. In 1993, the NRC approved operation of North Anna fuel to burnup levels consistent with these findings (References 6 and 7).

Increasing burnup has limited effect on the short-lived isotope inventory in the fuel due to the development of an equilibrium condition between production and decay. Instead, extended burnups increase the fraction of the short-lived isotopes that migrate into the fuel-clad gap region (Reference 15). With increasing burnup there is no decrease in fuel rod integrity or the leakage probability of a fuel rod. However, with the increased activity in the clad-gap region, increased burnup can result in increased activity being released into the reactor coolant under normal operations. The increase is small enough to be easily accommodated by the reactor coolant clean up system.

Changes to the bases for spent fuel pool criticality calculations (e.g., no Boraflex with soluble boron credit instead) has no environmental impact. These changes do not affect the operating power or fuel burnup. There are no new release paths or new types of effluents. The boron concentration will be maintained at a slightly higher value than is required by the current administrative limit, but the magnitude of any current release is not expected to change.

The operating power, average batch fuel burnup limits and lead rod burnup limits will not be changed for North Anna concurrent with the fuel enrichment increase or changes to the SFP criticality calculations. It is concluded that the increase in the maximum fuel enrichment to 4.6 weight percent and the changes to the SFP calculations for North Anna Units 1 and 2 will not have a significant adverse environmental effect.

4.6 Mechanical Integrity

4.6.1 Rod Integrity

Currently there is an administrative limit on boron concentration in the spent fuel pool. The current Technical Specification requirements proposed here are slightly higher. Therefore, fuel storage conditions are not expected to change significantly. In addition, boron is not known to react directly with the fuel cladding, so there is no expected impact on rod integrity due to the new Technical Specification on conditions in the spent fuel pool.

The integrity of the fuel rods is also not directly affected by the initial fuel enrichment. Rod integrity is more sensitive to operating power, radial peaking factors and operating temperature of the cladding than to fuel enrichment or burnup. Extensive studies have been conducted that show that there is no loss in fuel integrity for rod average burnups reaching and exceeding 60,000 MWD/MTU as long as the operating power remains normal (Reference 15).

The fuel melt temperature for unirradiated UO_2 is 5080°F. Although this property of UO_2 is not a function of enrichment, the fuel melt temperature decreases with burnup (Reference 14). The North Anna overpower limits are set to ensure no fuel melting occurs, and this is verified each cycle as part of the reload safety analysis.

The current generation North Anna fuel uses ZIRLO for fuel cladding, guide thimbles, instrumentation thimbles, and mixing vane grids. ZIRLO replaced Zircaloy for the purpose of enhancing the reliability of the clad and extending fuel burnup levels. However, References 14 and 15 deal with fuel assemblies whose cladding is constructed from Zircaloy.

The ZIRLO alloy is similar to Zircaloy-4 in chemical composition, physical properties and chemical properties. The ZIRLO alloy has improved corrosion resistance and dimensional stability under irradiation. Westinghouse has addressed the performance of the ZIRLO alloy in fuel assemblies in Reference 16. This assessment concludes that there is no degradation in the performance of the ZIRLO alloy to a burnup beyond the current lead rod burnup limits.

The operating power, radial peaking factors and operating temperatures will not be changed for North Anna when the fuel enrichment is increased. Therefore, it is concluded that the increase in the maximum fuel enrichment to 4.6 weight percent for North Anna Units 1 and 2 will not have a significant adverse effect on the integrity or performance of the fuel rods. This also indicates that the probability of leakage from the fuel rods under normal conditions will not be affected by the increased fuel enrichment.

4.6.2 Control Rod Insertion

Concerns have been raised about the possible effects of fuel burnup on the fuel assembly structure, potentially leading to conditions that could interfere with control rod insertion and thus the safe shutdown of the reactor. It has been postulated that such conditions could develop at higher fuel burnups as a 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. Note that changes to the spent fuel pool design basis will not significantly affect the fuel storage conditions, and so will not impact the mechanical condition of the fuel assembly or control rod insertion.

The irradiation induced factors that could conceivably 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 the North Anna fuel, with no increase in the fuel

burnup limits, will not adversely affect fuel performance in any way which would affect the ability of the control rods to fully insert.

Virginia Power has continued to perform cycle specific calculations to verify that our core designs incorporate sufficient shutdown margin to ensure safe shutdown if the most reactive RCCA fails to fully insert. An increase in the initial fuel enrichment from 4.3 to 4.6 weight percent of U^{235} or changes to the SFP design basis will not affect the nature of such shutdown margin calculations for North Anna Units 1 and 2.

4.7 Dry Fuel Storage

The changes to the spent fuel pool design basis will not affect the conditions or procedures for loading casks. The loading of casks requires 2000 ppm boron. This concentration may increase as casks are relicensed for higher burnups. However, this is not dependant on the current proposed Technical Specifications requirements.

There is no fuel enrichment limit specified in the North Anna 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 cask storage used at the North Anna ISFSI. It should be noted that the cask designs currently in use at North Anna are licensed for lower initial fuel enrichments and fuel assembly burnups than may be found in current generation fuel.

Virginia Power 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.

4.8 Assessment of Unreviewed Safety Questions

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

4.8.1 Probability and Consequences of Previously Evaluated Accidents

The proposed increase in maximum fuel enrichment or the changes to the SFP design basis will not increase the probability of an accident previously evaluated in the North Anna Units 1 and 2 UFSAR. The only accidents for which the probability of occurrence is potentially affected by the fuel enrichment and SFP changes involve criticality events during fuel handling and storage (e.g., fuel mispositioning). Positioning of fuel assemblies in the SFP has always been administratively controlled. The new Technical Specifications place additional restrictions on the determination of the appropriate position in the SFP for each fuel assembly, but these considerations are incorporated into our administrative tools and guidelines. The controls on fuel movement, including any checks to ensure the assembly is placed in the specified location, remain unchanged. Criticality analyses have been performed that demonstrate that the K_{eff} during the handling and storage of both new and spent fuel is low enough to ensure subcriticality during postulated accident conditions. In addition analyses of the dilution of the spent fuel pool have been performed to ensure that there is adequate time for a dilution accident to be found and mitigated before criticality is reached in the spent fuel pool. 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 7, radiological consequences of other accidents previously evaluated in the North Anna Units 1 and 2 UFSAR will not be increased.

4.8.2 Possibility of Accidents Not Previously Evaluated

The possibility of an accident which is different from any already discussed in the North Anna Units 1 and 2 UFSAR is not created. Although there are new restrictions on placement of fuel in the SFP, the controls on fuel movement to the administratively specified locations in the pool are unchanged. The higher enrichment fuel and the new Technical Specifications for the spent fuel pool do not require any new or different plant equipment, and do not change the manner in which currently installed equipment is operated. There are no changes to normal core operation, and the units 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. Existing safety analyses of record will remain applicable for use of fuel with the higher initial enrichment.

4.8.3 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 North Anna Units 1 and 2 UFSAR is not increased. The design of cores which incorporate fuel at the higher initial enrichment and the spent fuel pool 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, including fuel mispositioning and pool dilution scenarios. The new Technical Specifications on the spent fuel pool do not introduce any new effluents or release paths, and do not affect the magnitude of any currently analyzed releases. Also, since the burnup limit of the fuel is not being increased, the radiological consequences of any malfunction of equipment important to safety previously evaluated in the North Anna UFSAR is not increased by the use of a higher fuel enrichment limit.

4.8.4 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 North Anna Units 1 and 2 UFSAR is not created. The design for North Anna cycles which incorporate the higher enriched fuel will meet all applicable design criteria. Fuel handling and storage in the spent fuel pool with additional restrictions on the selection of fuel storage locations and increased boron concentration will also 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. No new or different plant equipment is introduced, and the operation of currently installed equipment is not changed. The use of higher

enriched fuel and the changes to the SFP design basis have the potential to affect only criticality events during fuel handling and storage. Safety analyses demonstrated that K_{eff} will remain sufficiently low to ensure subcriticality, so no new releases will result and there is no impact on radiological consequences of accidents.

4.8.5 Margin of Safety

The margin of safety as defined in the Bases to any North Anna Technical Specification is not reduced. Safety analyses of record will remain applicable for the operation of fuel with a higher initial U^{235} enrichment and changes to the spent fuel pool. Criticality analyses demonstrate that the limits on K_{eff} 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 North Anna Units 1 and 2 Technical Specifications 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 North Anna 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 or changes to the spent fuel pool design basis.

Based on the evaluations and analyses results presented in the foregoing safety significance evaluation, it has been demonstrated that increasing the North Anna Units 1 and 2 maximum initial fuel enrichment to 4.6 weight percent U^{235} and changing the design basis of the spent fuel pool to eliminate any credit for Boraflex but take credit for soluble boron in the pool 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.

Table 4.1
North Anna Fresh Fuel Storage Rack Design Data

Item	Dimension (inches)	Tolerance (inches)
Storage tube pitch	21.0 or 30.0	1.0
Cell I.D.	9.0	N/A
Storage tube wall thickness	0.125	0.01
Concrete wall thickness	9.0	N/A
Concrete wall thickness	72.0	N/A
Concrete floor thickness	12.0	N/A
Storage tube length	161.375	N/A

Table 4.2
North Anna Fuel Assembly Design Data

Design Input	Value
Fuel Rod Pitch	1.25984 cm
Clad Material	Zircaloy-4 or ZIRLO
Clad O.D.	0.94996 cm
Clad Thickness	0.05715 cm
Pellet O.D.	0.81915 cm
Dish Fraction	1.2 %
Pellet Material	95 % Theoretical Density of UO_2*
Guide Tube Material	Zircaloy-4 or ZIRLO
Guide Tube O.D. – Above Dashpot	1.22428 cm (LOPAR) 1.20396 cm (NAIF)
Guide Tube Thickness – Above Dashpot	0.04064 cm
Instrumentation Thimble Material	Zircaloy-4 or ZIRLO
IT Thickness	0.04064 cm
IT Inner Diameter	1.143 cm (LOPAR) 1.12268 cm (NAIF)
Active Fuel Length	365.76 cm
Fuel Rods/Assembly	264
Guide Tubes & Instrument Tubes/Assembly	25
Assembly Pitch	21.50364 cm

* A nominal pellet density of 95.5% was used in the analysis. This is consistent with recent fuel batches and is conservative versus 95% based on sensitivity cases.

Table 4.3
KENO K-effective vs. Moderator Density
17x17 Westinghouse Fuel, 4.6 w/o LEU, Nominal Design Data

(Water Density = 0.9982 g/cc @ Volume Fraction = 1.0)

Moderator Volume Fraction	K-effective	Standard Deviation
1.0	0.88083	0.00042
0.8	0.79117	0.00041
0.6	0.68771	0.00039
0.4	0.59919	0.00037
0.3	0.59762	0.00034
0.2	0.67585	0.00034
0.1	0.86435	0.00036
0.07	0.90516	0.00035
0.065	0.90726	0.00037
0.06	0.90660	0.00037
0.055	0.90356	0.00036
0.05	0.89715	0.00035
0.03	0.82256	0.00034
0.001	0.55250	0.00029

Table 4.4
KENO Tolerance Case Results
Fresh Fuel Storage Rack
17x17 Westinghouse Fuel, 4.6 w/o LEU, Nominal Design Data

Uncertainty and Bias

Item	K-effective	Standard Deviation	Delta-K
Tolerance Reference[a]	0.93994	0.00037	0.000611
+0.5% Enrichment	0.94266	0.00034	0.003549
97.5% TD, 0% dish	0.94375	0.00036	0.004662
Conservative Geometry	0.96353	0.00035	0.24430
Methodology Uncertainty			0.000990
Total Uncertainty			**0.025150[b]**
Method Bias			0.00357
Total Uncertainty and Bias			**0.02872**

Maximum K_{eff} including Uncertainty and Bias

Total Uncertainty and Bias	**0.02872**
Nominal Keffective	**0.90726**
Maximum Keff	**0.93598[c]**

[a]Tolerence Reference case uses a simplified KENO model relative to that used to determine nominal K_{eff}

[b]Root mean square of individual uncertainty Delta-K

[c]Total uncertainties + Total bias + Nominal K_{eff}

Table 4.5
KENO Tolerance Results
Spent Fuel Storage Rack
2.0 w/o LEU, 0 ppm Boron, 3x3

Uncertainty

Item	K-effective	Standard Deviation	Delta-K
Reference	0.96494	0.00021	0.000347
+0.5% enrichment	0.97243	0.00020	0.007969
+2% density	0.96803	0.00022	0.003592
0% dishing	0.96680	0.00021	0.002350
-0.005 wrapper	0.96734	0.00022	0.002902
-0.005 cell wall thickness	0.96753	0.00021	0.003080
Rack pitch base case	0.96743	0.00020	
Rack pitch –1/64	0.96956	0.00020	0.002597
Rack pitch –1/8 1 row xy	0.96923	0.00020	0.002267
Fuel pitch, 10.27 inch pitch	0.97018	0.00021	0.003229
Methodology Uncertainty			0.000990
Total Uncertainty			**0.010867**[a]

Bias

Item	K-effective	Standard Deviation	Delta-K
Reference case with correct width wrappers	0.96602	0.00021	0.00108
Method bias			0.00357
Temp bias			0.01300
Wrapper error bias			0.00108
Total Bias			**0.01873**

Maximum K_{eff} including Uncertainty and Bias

Total Uncertainty and Bias	**0.02960**
Nominal Keffective	**0.96494**
Maximum Keff	**0.99454**[b]

[a]Root mean square of individual uncertainty Delta-K

[b]Total uncertainties + Total bias + Nominal K_{eff}

Table 4.6
KENO Tolerance Results
Spent Fuel Storage Rack
20 x1.56 w/o LEU, 4x4.60 w/o LEU, 1 empty, 0 ppm Boron, 5x5

Uncertainty

Item	K-effective	Standard Deviation	Delta-K
Reference	0.97701	0.00021	0.000347
+0.5% enrichment	0.98099	0.00022	0.004482
+2% density	0.97922	0.00022	0.002712
0% dishing	0.97829	0.00023	0.001794
-0.005 wrapper	0.97887	0.00022	0.002362
-0.005 cell wall thickness	0.97902	0.00022	0.002512
Rack pitch –1/64	0.97715	0.00022	0.000642
Rack pitch –1/8 1 row xy	0.98035	0.00022	0.003842
Fuel pitch, 95/95	0.98056	0.00022	0.004052
Methodology Uncertainty			0.000990
Total Uncertainty			**0.008650**[a]

Bias

Item	K-effective	Standard Deviation	Delta-K
Method bias			0.00357
Temp bias			0.01000
Total Bias			**0.01357**

Maximum K_{eff} including Uncertainty and Bias

Total Uncertainty and Bias	0.02222
Nominal Keffective	0.97701
Maximum Keff	**0.99923**[b]

[a]Root mean square of individual uncertainty Delta-K

[b]Total uncertainties + Total bias + Nominal K_{eff}

Table 4.7
KENO Tolerance Results
Spent Fuel Storage Rack
20 x1.56 w/o LEU, 4x4.60 w/o LEU, 1 empty, 230 ppm Boron, 5x5

Uncertainty

Item	K-effective	Standard Deviation	Delta-K
Reference	0.92593	0.00023	0.000380
+0.5% enrichment	0.93052	0.00022	0.005115
+2% density	0.92917	0.00021	0.003754
0% dishing	0.92757	0.00022	0.002165
-0.005 wrapper	0.92675	0.00021	0.001334
-0.005 cell wall thickness	0.92797	0.00023	0.002577
Rack pitch –1/64	0.92729	0.00022	0.001885
Rack pitch –1/8 1 row xy	0.92990	0.00022	0.004495
Fuel pitch, 95/95	0.92954	0.00021	0.004124
Methodology Uncertainty			0.000990
Total Uncertainty			0.009576[a]

Bias

Item	K-effective	Standard Deviation	Delta-K
Method bias			0.00357
Temp bias			0.01000
Total Bias			0.01357

Maximum K_{eff} including Uncertainty and Bias

Total Uncertainty and Bias	0.02315
Nominal Keffective	0.92593
Maximum Keff	0.94908[b]

[a]Root mean square of individual uncertainty Delta-K

[b]Total uncertainties + Total bias + Nominal K_{eff}

Table 4.8
KENO Tolerance Case Results
Fuel Misload
20x1.56 w/o LEU, 4x4.60 w/o LEU

# of Misloaded Assemblies	Boron [ppm]	K-effective	K-effective + Uncertainty	Standard Deviation
0	230	0.92593	0.94908	0.00023
1	780	0.92044	0.94359	0.00022

Figure 3.1
North Anna Burnup Credit Requirements for Non-Matrix Storage Configuration
(New T.S. Figure 3.7.15-1)

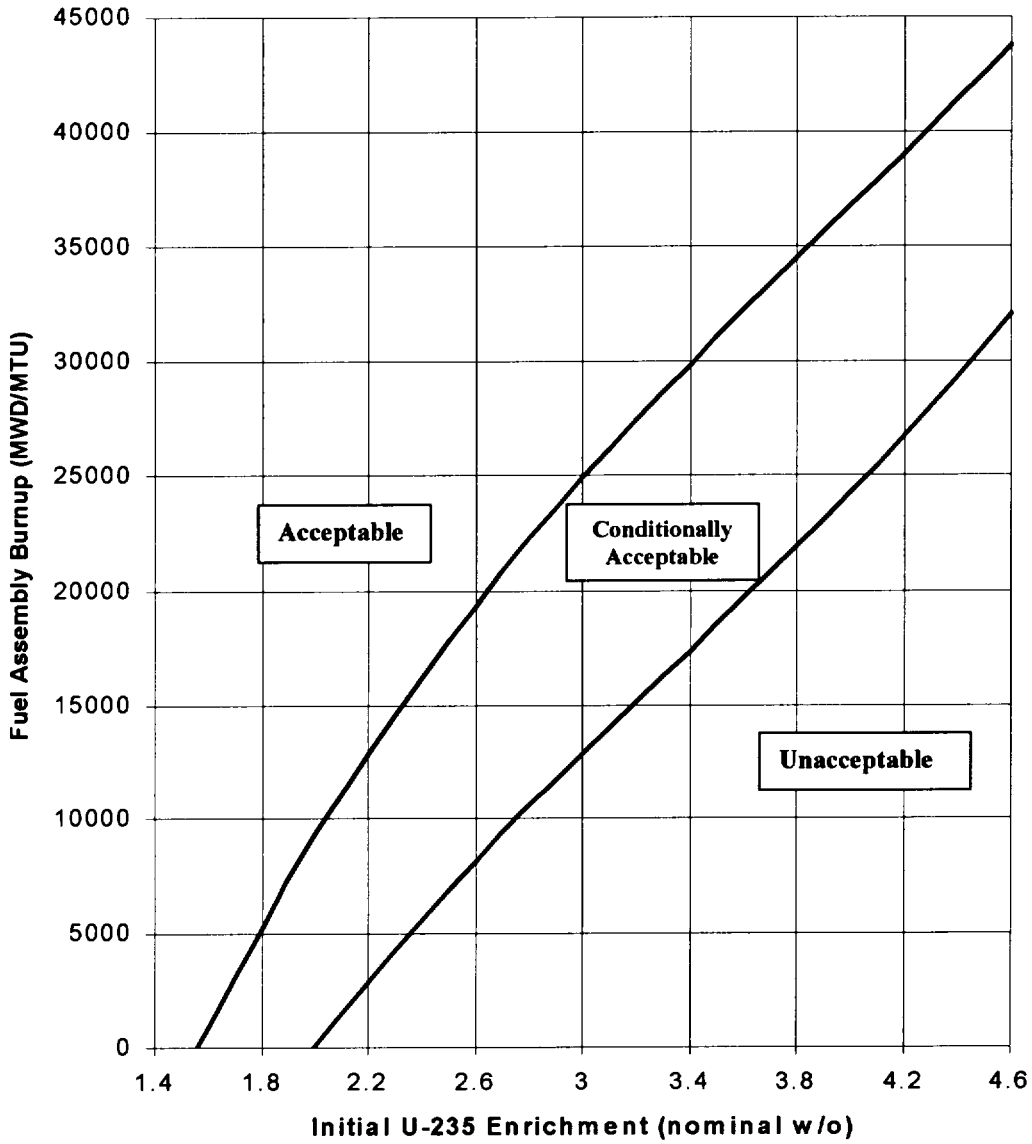


Figure 4.1
North Anna Fresh Fuel Storage rack Layout
KENO Model Plot

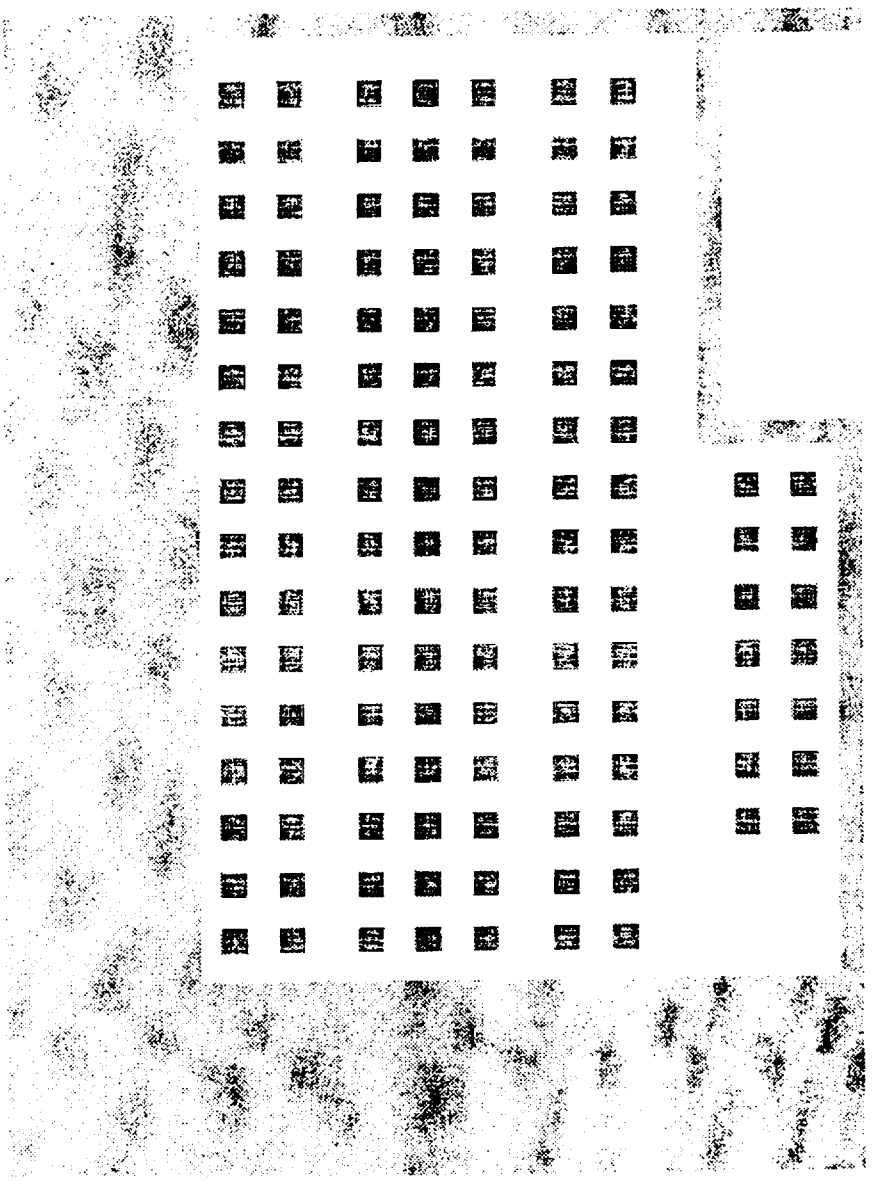
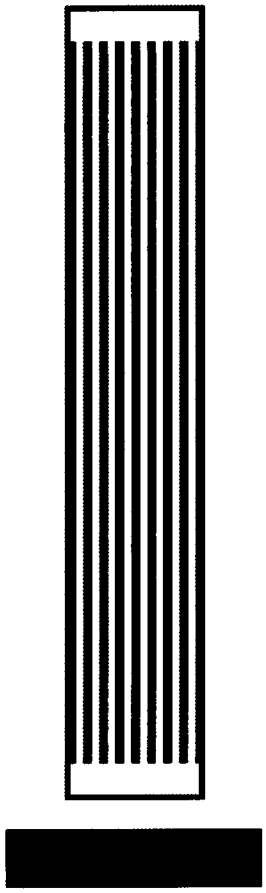


Figure 4.2
North Anna Fresh Fuel Storage Rack Layout
Axial Model



Moderator above storage tube (assume flooded room 300 cm)

Fuel length 144 inches**

Storage tube length 13 feet 5.375 inch (DWG 11715-FV-18A-6) excluding 4 inch top lead-in

Fuel starts at floor + 0.375 inch (DWG 11715-FV-18A) + 3.278 inch**

0.75 inch gap (assume water, DWG 11715-FV-18A-6)

12 inch concrete floor (DWG 11715-FC-27E-6)

Assume 300 cm moderator in room below (DWG 11715-FM-3A)

** - Typical fuel dimensions from Westinghouse letter 93VP*-G-0022, June 1993

Figure 4.3
North Anna Fresh Fuel Storage
K-effective versus Water Density, 4.6 w/o 17x17 Fuel

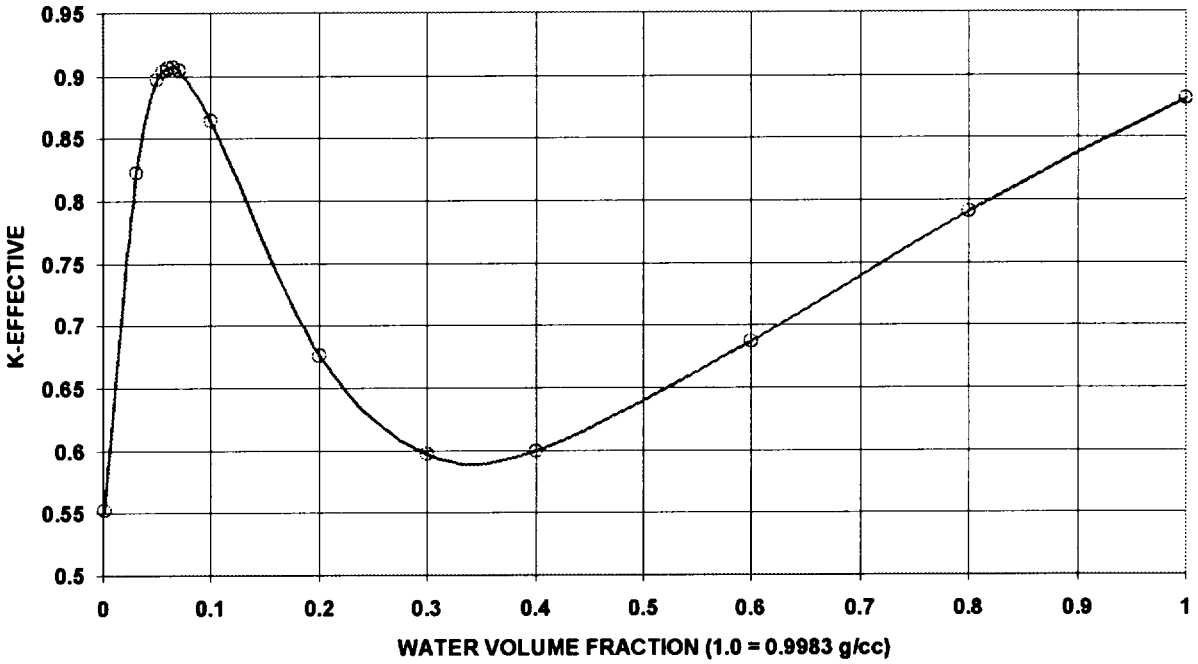
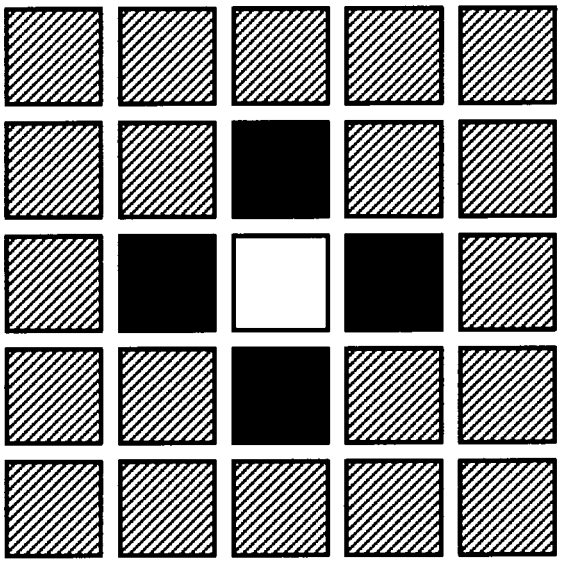
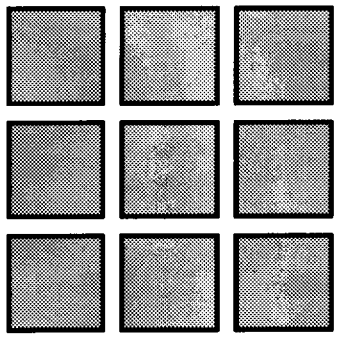


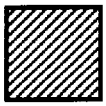
Figure 4.4
Spent Fuel Storage Matrix Regions



5X5 Matrix

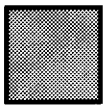


Non Matrixed

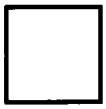


Low reactivity location (max reactivity equivalent to 1.56 w/o U^{235} fresh fuel)

High reactivity location (e.g. for fresh and low burnup once-burned fuel; max reactivity equivalent to 4.6 w/o fresh fuel)



Intermediate reactivity location (max reactivity equivalent to 2.0 w/o U^{235} fresh fuel)



Storage cell with no fuel assembly

Figure 4.5
Axial Burnup Reactivity Bias
3-D Burnup Distribution vs. 2-D (Axially Infinite) Uniform Burnup

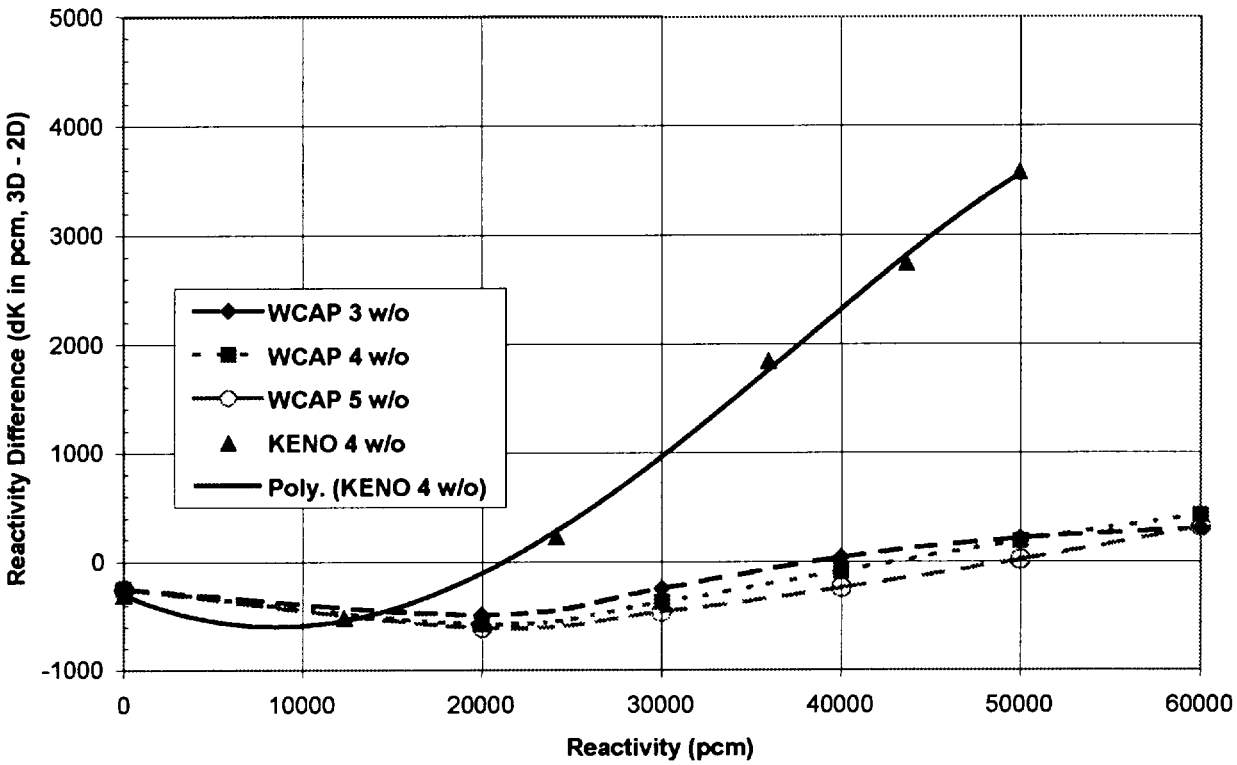
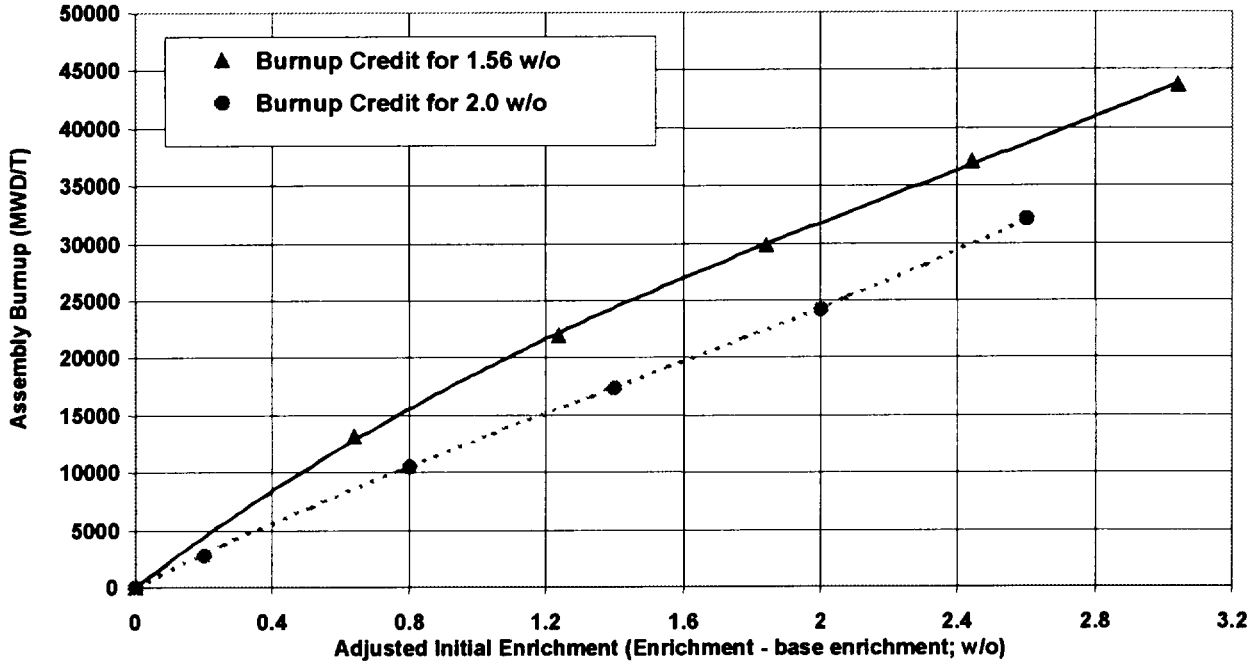


Figure 4.6
Burnup Credit Requirement
1.56 w/o and 2.0 w/o Equivalent
Westinghouse 17x17 LEU



5.0 CONCLUSIONS

Virginia Power proposes modifying the basis of the spent fuel storage pool racks to eliminate any credit for Boraflex in the SFP criticality calculations. This will require establishing Technical Specifications controlling the soluble boron concentration and the placement of fuel assemblies in the spent fuel pool. In addition, Virginia Power proposes to increase the maximum fuel enrichment for U^{235} for North Anna Units 1 and 2 from the current Technical Specifications limit of 4.3 weight percent U^{235} (maximum) to 4.6 weight percent U^{235} (nominal). 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 address the impact on the new and spent fuel storage areas, which are common to North Anna Units 1 and 2.

The criticality analysis for the fresh fuel storage racks demonstrated that the North Anna new fuel storage area meets the criticality limit of K_{eff} less than 0.98 with the higher fuel enrichment, and is safe under the criticality specifications set forth in Section 9.1.1 of the Standard Review Plan (NUREG-0800).

The North Anna spent fuel racks were evaluated assuming no credit for Boraflex but taking credit for soluble boron using NRC approved Westinghouse Owners Group methodology. It was demonstrated that the K_{eff} in the North Anna spent fuel pool will remain less than 1.0 assuming no soluble boron in the pool, and will remain less than 0.95 for all other conditions, including postulated accident scenarios, for fuel enriched to a maximum of 4.6 w/o U^{235}.

Use of a higher initial fuel enrichment in North Anna fuel assemblies or changes to the spent fuel pool design basis will not affect the inputs to any North Anna safety analysis, nor will the consequences of accident scenarios described in the North Anna 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, the elimination of Boraflex and the use of soluble boron credit in the spent fuel pool criticality analysis will not result in an unreviewed safety question as defined by 10 CFR 50.59.

6.0 REFERENCES

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5. Letter from Leon B. Engle (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.

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

7. Letter from Leon B. Engle and Bart C. Buckley (U.S. Nuclear Regulatory Commission) to W. L. Stewart (Virginia Electric and Power), "Surry, Units 1 and 2, and North Anna, Units 1 and 2 – Removal of 45,000 MWD/MTU Batch Average Burnup Restriction", April 20, 1994.

8. "Standard Review Plan", NUREG-0800, Rev. 2, July, 1981

9. "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", WCAP-14416-NP-A, Rev. 1, November, 1996.

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11. "Review of Axial Burnup Distribution Considerations for Burnup Credit Calculations", ORNL/TM-1999/246, March, 2000.

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13. Letter from B. D. McKenzie (Westinghouse Electric Corporation) to J. M. Dowhan (Virginia Electric and Power), "Surry and North Anna Power Stations, Fuel performance RSAC Data", October 21, 1991.

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15. 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.

16. Davidson, S. L., and Ryan, T. L., "VANTAGE+ Fuel Assembly Reference Core Report", WCAP-12610-P-A, April, 1995.

17. Grimes, B. K., "Reactivity Insertion Transient and Accident Limits for High Burnup Fuel", NRC Information Notice 94-64, Supplement 1, April 6, 1995.

Appendix A

Spent Fuel Pool Boron Dilution Analysis

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

ANALYSIS OF SPENT FUEL POOL DILUTION

1.0 INTRODUCTION

This documents the dilution assessment which is a required component for crediting boron in the North Anna spent fuel pool (SFP) rack criticality analysis. The NRC SER (Reference 1) for the Westinghouse Owners Group spent fuel pool criticality analysis requires each licensee to perform a plant-specific analysis of the potential for spent fuel pool dilution to the minimum concentration required to maintain K_{eff} less than 0.95. This analysis ensures that sufficient time is available to detect and mitigate the dilution before the spent fuel rack criticality analysis 0.95 K_{eff} design basis is reached. This analysis must consider dilution pathways and flowrates and other key plant-specific features such as spent fuel pool volume. The analysis results include the time and total volume of water associated with the dilution scenario identified.

The boron dilution analysis includes an evaluation of the following plant specific features:

- Dilution Sources and Flowrates
- Boration Sources
- Instrumentation
- Administrative Procedures
- Piping
- Loss of Offsite Power Impact
- Boron Dilution Initiating Events
- Boron Dilution Times and Volumes

It should be noted that even though potential dilution pathways and their associated flowrates are identified for the purpose of calculating dilution times and volumes, a major conservatism is inherent in the calculations. As indicated in Section 3.0, all but one (cavity seal failure) of the dilution scenarios involve only water addition to the spent fuel pool. Continuous water addition will result in pool overfill and spillage into the Fuel Building upon addition of only a small fraction of the water necessary to lose criticality control. It is considered incredible that the quantity of water identified in the dilution scenario of Section 3.0 could be added to the spent fuel pool without detection of its effects in the Fuel Building, Auxiliary Building or adjacent areas of the station.

2.0 SPENT FUEL POOL AND RELATED SYSTEM FEATURES

This section provides background information on the spent fuel pool and its related systems and features.

2.1 Spent Fuel Pool

The design purpose of the SFP is to provide safe storage of irradiated fuel assemblies. The pool is filled with borated water. The water functions to remove decay heat, provide shielding for personnel handling the fuel, and to reduce the amount of radioactive gases released during a fuel handling accident. Pool water evaporation takes place on a continuous basis, requiring periodic makeup. The makeup source can be unborated water. Since the evaporation process does not remove boron, evaporation actually increases the boron concentration in the pool.

The North Anna SFP is a reinforced concrete structure with a minimum 0.25 inch stainless steel liner. The pool structure is designed to meet seismic requirements. The pool is approximately 42 feet deep. The spent fuel pool at North Anna is shared by both units. Figure 2-1 shows a general layout of the pool. The pool contains two areas, a small section used solely for spent fuel cask loading and a larger section containing racks for storage of spent fuel. The smaller section is separated from the main pool by a 3-1/2-foot thick dividing wall. It is not possible to isolate the cask loading section from the fuel storage area due to holes cut in this dividing wall.

The North Anna spent fuel pool has a small pool section that is presently used solely for cask handling and will continue to have this function in the future. As mentioned, it is not possible to isolate the cask loading section from the main pool. However, there are no racks for fuel storage in the cask handling area. Therefore, only the volume of the large pool section is included in this analysis in order to provide a more conservative volume for the dilution assessment. This volume contains 409,695 gallons when filled to the level associated with the pool low level alarm.

2.2 Spent Fuel Storage Racks

The spent fuel storage racks consist of double walled, stainless steel boxes welded to each other maintaining a cell pitch of 10-9/16 inches. Each storage cell has an interior height of 168 inches. The spent fuel racks are designed to support and protect the spent fuel assemblies under normal and credible accident conditions. Their structural strength ensures the ability to withstand combinations of dead loads, live loads (fuel assemblies) and safe shutdown earthquake loads.

2.3 Fuel Pit Cooling and Refueling Purification System

The fuel pit cooling and refueling purification system removes residual heat from spent fuel stored in the SFP. This system is also used to purify and maintain the optical clarity of the spent fuel pool water and the water in the reactor refueling cavity and the refueling water storage tanks.

The system is common to both units and normally serves only the SFP. Although shared by both units, the safety function is not affected by an accident in either unit. The system is capable of handling a maximum heat load corresponding to a complete core offload (planned or unplanned) commencing 150 hours after shutdown.

The fuel pit cooling piping and equipment are designed to Seismic Class I requirements. Portions of the refueling purification piping and equipment are not designed to Seismic Class I requirements. Operating procedures provide contingency actions to assure that the design basis of adjacent systems are maintained in the event of a seismic event with the refueling purification system aligned to a Seismic Class I system.

The portion of the fuel pit cooling and refueling purification system used to cool the spent fuel pool water has 2 shell and tube heat exchangers and 2 circulating pumps (2750 GPM capacity each). The heat exchangers and pumps are arranged for cross connection if necessary.

All spent fuel pool penetrations are located so that at least 23 ft 1 in of water would remain above the active portions of the spent fuel assemblies stored in the spent fuel pool even if water should drain through the penetrations, thus ensuring adequate shielding for the spent fuel assemblies. The system also includes 3 refueling purification pumps (400 GPM capacity each), 2 filters and an ion exchanger.

Spent fuel pool water can be purified by pumping a portion of the fuel pit cooling loop flow through the refueling purification filters and ion exchanger with the refueling purification pumps. The filters and ion exchanger are operated in series. The refueling purification pumps can be run to purify the spent fuel pool water independently of the cooling pump operation. The water surface of the spent fuel pool is maintained free of floating material by 2 permanently installed skimmers connected to the suction of the spent fuel pool cooling pumps.

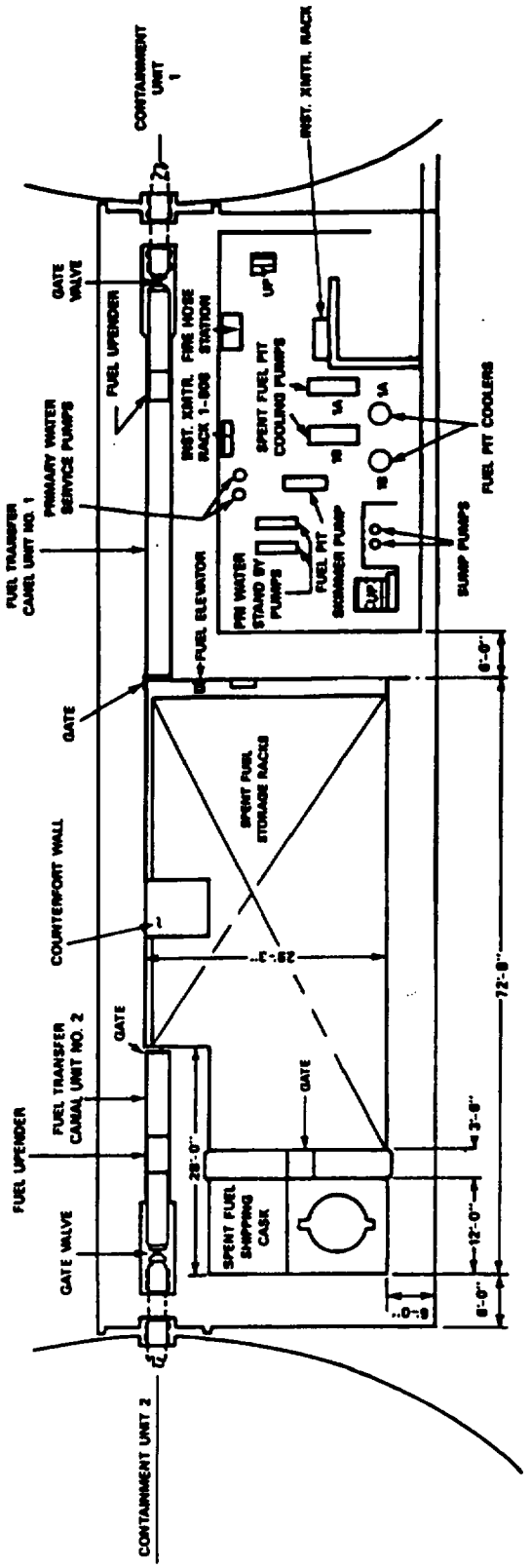

Figure 2-1: North Anna Spent Fuel Pool Layout

2.4 Dilution Sources

2.4.1 Chemical and Volume Control System (Makeup Water)

During normal operation, evaporative losses require that makeup be provided as often as once or twice a week depending on ambient conditions. Normal makeup to the SFP is via one of two chemical and volume control systems (Unit 1 or 2). Both systems are similar in regards to the makeup path from the blender to the spent fuel pool. The discussion below is for Unit 1, but is applicable to either system.

Makeup from the CVCS system is through a 2 inch line on the downstream side of the boric acid blender and is isolated by 2 manual valves (1-CH-230 and 1-CH-232). 1-CH-230 is a normally closed valve that isolates a line that supplies the spent fuel pool system along with the RWST through a check valve. 1-CH-232 is a normally closed valve that specifically isolates the spent fuel pool cooling system from the CVCS. Thus, there are two normally closed manual valves that must be opened and left open to allow a dilution event via the CVCS. This could result in a flowrate of up to 200 gpm to the spent fuel pool. Since North Anna has two units, there are two blenders available for makeup to the SFP.

2.4.2 Dilution from Open Demineralizer Valves

The primary grade (PG) water system connects directly to the spent fuel pool cooling system at the purification loop demineralizer through a 2 inch line that is isolated by two normally closed manual valves, 1-RP-70 and 1-LW-749. The connection is normally used to sluice and refill the demineralizer during a resin changeout. The demineralizer has two additional connections. The first is a resin flush line that is connected through a normally closed manual valve, 1-RP-72, to a 2 inch spent resin header which in turn connects to the spent resin tank through a second normally closed manual isolation valve, 1-LW-542. The second is a resin fill line that is a flanged connection and is only opened when adding new resin to the demineralizer.

The resins in the demineralizer tank are flushed or changed one to three times a year depending on the number of refueling outages. During this operation, the demineralizer tank is isolated and the valve to provide PG water for flushing is opened. A dilution path to the spent fuel pool is created after the operation is completed, if the demin tank is unisolated (no error), but the wrong valves are closed to isolate the PG water. The dilution flowrate from the demin tank is conservatively estimated to be 200 gpm.

2.4.3 Component Cooling Water

Component cooling water (CCW) is the cooling medium for the SFP cooling system heat exchangers. There is no direct connection between the component cooling system and the SFP cooling system. However, if a leak were to develop in a heat exchanger that is in service, the connection would be made. In case of a leak, the CCW water would be expected to leak into the SFP cooling system because the CCW system normally operates at a slightly higher pressure that the SFP cooling system. A conservative flowrate of 100 gpm is estimated for the leakage.

2.4.4 Primary Grade Water

There are three PG water hose connections in the vicinity of the pool, only one of which is directly next to the pool. The other two point away from the pool and if left open, the water would flow into the pipe trench and not into the pool. Misalignment of the 3/4 inch PG lines around the pool are not analyzed further in this section since it is considered unlikely that an open valve with water flowing onto the floor would go undetected. Additionally, the dilution flow rate (35 gpm) is sufficiently low that it would take numerous operator rounds to not detect it before reaching the criticality limit.

2.4.5 Fire Protection

For North Anna, there is an emergency fill line from the underground fire main that may be used in the event of a loss of water in the SFP and a loss of the normal makeup from either unit CVCS system. This line is isolated from the fire main outside the fuel building by a post indicating gate valve (1-FP-56). This valve is also locked closed and is painted gray rather than the normal red color for fire protection valves and pipes. The pipe is normally empty, so a random break of this pipe was not considered.

However, the valve is cycled once a year as part of the normal surveillance for cycling the fire protection valves. The cycling of this valve during the annual surveillance consists of closing an upstream valve, then opening the emergency fill valve and then closing it. The upstream valve is then opened and the emergency fill valve is verified not to leak by. If the valve was inadvertently opened, 400 gpm of fire water would flow into the SFP.

There is also a 10 foot section of fire protection piping in the vicinity of the spent fuel pool. A rupture of this section of fire protection piping would release less than 350 gpm.

2.4.6 Plant Heating System Steam Lines

In the vicinity of the spent fuel pool there are 2 inch and 1.5 inch sections of plant heating system steam lines. It is assumed, for conservatism, that a plant heating steam line rupture would result in all condensed steam collecting in the spent fuel pool. The flow rate for a plant heating steam line rupture would be less than 35 gpm.

2.4.7 Dilution Source and Flow Rate Summary

Based on the evaluation of potential spent fuel pool dilution sources summarized above, the following dilution sources were determined to be capable of providing a significant amount of non-borated water to the spent fuel pool. The potential for these sources to dilute the spent fuel pool boron concentration down to the design basis boron concentration (1200 ppm) will be evaluated in Section 3.0.

SOURCE	APPROXIMATE FLOW RATE (GPM)	SECTION
Chemical and Volume Control System		
- Makeup Valve Open	200	2.4.1
Primary Grade Water System		
- Open Demineralizer Valve	200	2.4.2
- Hose Connections	35	2.4.4
	35 (pipe break)	2.4.4
Component Cooling Water	100	2.4.3
Fire Protection		
- Emergency SFP Fill Valve	400	2.4.5
	350 (pipe break)	2.4.5
Plant Heating System Steam Line	35 (pipe break)	2.4.6

2.5 Boration Sources

The nominal source of borated water to the SFP is through the blender in the CVCS system. It is also possible to borate the spent fuel pool by the addition of dry boric acid directly into the spent fuel pool.

2.5.1 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) is the normal borated makeup source for the spent fuel pool. The CVCS blender is connected to the spent fuel pool cooling system by a two inch line. This connection is used to supply water at a specific boron concentration to the pool. Concentrated boric acid is supplied to the CVCS blender from the boric acid storage tanks via the boric acid transfer pumps. Makeup water is supplied to the blender by the primary grade water system. The rate of makeup addition through the blender is estimated to be 200 gpm. The supply from the blender to the spent fuel pool cooling system can have a boron concentration of anywhere from 0 to the boric acid storage tank concentration depending on the control setting of the blender.

2.5.2 Direct Addition of Boric Acid

If necessary, the boron concentration of the spent fuel pool can be increased by emptying bags of dry boric acid directly into the spent fuel pool. The dry boric acid will dissolve into the spent fuel pool water and will be mixed throughout the pool by the spent fuel pool cooling system flow and by the thermal convection created by the spent fuel decay heat.

2.6 Spent Fuel Pool Instrumentation

Instrumentation is available to monitor spent fuel pool water level and temperature, and the radiation levels in the spent fuel pool enclosure. Additional instrumentation is provided to monitor the pressure, flow and temperature of the spent fuel pool cooling and purification system.

The instrumentation provided to monitor the temperature of the water in the SFP is locally indicated as well as annunciated in the control room. The water level instrumentation alarms, high and low, are annunciated in the control room. The instrumentation which monitors radiation levels in the SFP enclosure, provides high radiation alarms locally in the SFP enclosure and in the control room.

A change of six inches in spent fuel pool level requires approximately 6700 gallons of water. If the pool level was raised from the low level alarm point (6 inches below normal) to the high level alarm (6 inches above normal) a dilution of approximately 13400 gallons could occur before an alarm would be received in the control room. If the spent fuel pool boron concentration were at 2200 ppm initially, such a dilution would only result in a reduction of the pool boron concentration of approximately 70 ppm.

2.7 Administrative Procedures

The following administrative controls are in place to control the spent fuel pool boron concentration and water inventory:

1. Procedures are available to aid in the identification and termination of dilution events.

2. The procedures for loss of inventory (other than evaporation) specify that borated makeup sources be used as makeup sources. The procedures specify that non-borated sources only be used as a last resort.

3. In accordance with procedures, plant personnel perform rounds in the SFP enclosure once every six hours. The personnel making rounds to the SFP are trained to be aware of a change in status of the SFP. They are instructed to check the temperature and the level of the pool and conditions around the pool during plant rounds.

4. Administrative controls are placed on some of the potential dilution paths.

5. An administrative procedure is in place that requires the spent fuel pool boron concentration be verified at least every seven days.

North Anna plant procedures have been reviewed to ensure that adequate administrative controls exist for potential dilution pathways and that the spent fuel pool conditions are monitored on a regular basis. The North Anna procedures for makeup to the pool are detailed and contain steps for independent verifications to ensure that the pool is not diluted and to confirm that the system lineup is returned to normal. This is also true for the procedures for operating the Refueling Purification (RP) system and the RP ion exchanger. The water in the pool is sampled once per week and verified to have a boron concentration greater than the administrative control limit of 2300 ppm at all times. Additionally, operators make rounds through the fuel building where the SFP is located every 6 hours. Therefore a pool dilution event that could cause the pool level to increase would be detected within six hours if the level alarm should fail. With the large pool

volume at North Anna, the dilution time to criticality for most of the dilution sources is several times greater than the frequency of operator rounds.

2.8 Piping

The piping located within the vicinity of the spent fuel pool consists of 2" fire protection lines, ¾" primary grade water lines and 2" and 1.5" plant heating systems lines. Of these, only the fire protection lines are seismically qualified.

2.9 Loss of Offsite Power Impact

Of the dilution sources listed in section 2.4 only the fire protection and the plant heating systems are capable of providing non-borated water to the spent fuel pool during a loss of offsite power.

One train of the spent fuel pool level alarms is powered by a vital bus in the event of a loss of offsite power.

The loss of offsite power would affect the ability to respond to a dilution event. The normal source of borated water (CVCS) to the SFP would not be available upon a loss offsite power. The RWST is also unavailable. Manual addition of dry boric acid to the pool could be used if it became necessary to increase the SFP boron concentration during a loss of offsite power.

3.0 SPENT FUEL POOL DILUTION EVALUATION

This dilution analysis supports a Technical Specifications change request that will increase the spent fuel pool boron concentration to 2500 ppm. The following calculation of dilution times and volumes is based on a previous evaluation assuming a spent fuel pool boron concentration limit of 2300 ppm. Since the previous dilution evaluation uses a more conservative limit than the requested boron concentration limit, the evaluation is not repeated for 2500 ppm.

3.1 Calculation of Boron Dilution Times and Volumes

Based on the North Anna spent fuel pool criticality analysis, the soluble boron concentration required for criticality requirements (i.e., a $K_{eff} < 0.95$) is estimated to be 900 ppm. This soluble boron concentration of 900 ppm is based on the "worst case" single misload accident in the spent fuel pool. The dilution analysis was conducted for an ending spent fuel pool boron concentration of 1200 ppm, which makes the dilution analysis more conservative by 300 ppm. The endpoints, 2300 ppm and 1200 ppm, were considered for the deterministic evaluation of dilution volume and time; thus a (2300 - 1200), or 1100 ppm boron dilution event was considered.

The dilution volumes and times for these scenarios are calculated based on the following equation:

$$t_{end} = \ln (C_o / C_{end})V/Q \qquad \text{(Equation 1)}$$

Where:

C_o is the boron concentration of the pool volume at the beginning of the event

C_{end} = the boron endpoint concentration

Q = dilution rate (gallons of water/minute)

V = volume (gallons) of spent fuel pool.

Equation 1 incorporates an assumption of perfect mixing between the dilution flow and the borated water in the spent fuel pool. The dilution sources identified (and potential letdown pathways) enter and leave the spent fuel pool at elevations near or at the water level. The present evaluation has concluded that the potential reactivity effects of boron concentration stratification are conservatively modeled by the perfect mixing assumption for the following reasons: 1) dilution flow entering the top of the pool passes through a significant amount of borated water (approximately 25 feet) prior to reaching the top of the fuel assemblies; 2) there are insignificant density differences (less than 2% for 32F to 140F water) between the dilution water and spent fuel water; 3) spent fuel decay heat will tend to establish a convective, axial mixing flow regime in the pool. During a dilution scenario, the consequence of these processes is that Equation 1 will

provide a conservative (smaller concentration) than actually expected in the vicinity of the spent fuel racks.

However, if mixing was not adequate, a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the spent fuel pool criticality analysis which shows that the spent fuel pool K_{eff} will be less than 1.0 on a 95/95 basis with the spent fuel pool filled with non-borated water. Thus even if a pocket of non-borated water formed in the spent fuel pool, Keff would not be expected to equal or exceed 1.0 anywhere in the pool.

The time to dilute depends on the initial volume of the pool and the postulated rate of dilution. The North Anna Power Station has a shared pool for the two units. There are two areas in the spent fuel pool, a small section used solely for spent fuel cask loading and a larger section containing racks for storage of spent fuel. The smaller section is separated from the main pool by a 3-1/2-foot thick dividing wall. It is not possible to isolate the cask loading section from the fuel storage area. To conservatively determine dilution times the volume of the cask area and transfer canals are ignored leaving only the volume associated with spent fuel storage. The effective pool volume for dilution has been calculated by assuming: 1) the pool level is at the minimum allowed value; 2) each of the 1737 spent fuel cells contains a fuel assembly. Subtracting the volume occupied by the spent fuel racks and assemblies, resulted in a pool volume of 409,695 gallons for use in calculation of dilution times.

Equation 1 is used to calculate the dilution time for a range of dilution rates from 20 gpm to 500 gpm for the spent fuel volume given above. Table 3-1 provides the North Anna dilution time and volume data for a dilution event from 2300 ppm to 1200 ppm. It should be noted that there is a major conservatism inherent in use of Equation 1 for modeling dilution: the dilution source flow introduced to the pool is assumed to be offset by pool letdown or overflow. This assumption is only true if there is a break or leak in a SFP cooling line or RP line while makeup to the pool is being performed. This scenario of a leak or break that results in a letdown equal to the makeup flow occurring at the same time as makeup to the pool is initiated is not considered credible. For the major potential incidents in which water is added to the pool, overflow into the Fuel Building is the only mechanism for displacement. It is non-credible that the quantity of water identified in the dilution scenario of Table 3-1 (266,542 gallons) could be added without detection of its effects in the Fuel Building, Auxiliary Building or other adjacent areas of the station.

To better assess the magnitude of dilution involved, the following calculation estimates the water volume required to be added to the spent fuel pool to cause pool overfill from six inches below the normal level:

Volume of water to Overfill Pool = Pool Area x Height x 7.48 gallons/cuft
= 1254.1 x 2.5 x 7.48
= 23,452 gallons

Percent of total volume = (23,452/266,542) x 100%
= 8.8%

This calculation demonstrates that upon the addition of less than 10% of the total water volume required to lose criticality control, the spent fuel pool will overflow. Continued water addition will create significant effects, detectable both inside and outside the Fuel Building.

TABLE 3-1
North Anna Dilution Time & Volume Data
Dilution Event: 2300 ppm to 1200 ppm

Dilution Flow Rate	Pool Volume 409,695 gal	
	Dilution Time (hr)	Dilution Vol (gal)
20 gpm	222	266,542
35 gpm	127	
50 gpm	89	
65 gpm	68	
100 gpm	44	
200 gpm	22	
250 gpm	18	
350 gpm	13	
500 gpm	9	

3.2 Evaluation of Boron Dilution Events

The potential spent fuel pool dilution events that could occur at North Anna are evaluated below:

3.2.1 Dilution Resulting from a CCW Leak

The assumption of a 100 gpm leak rate (Section 2.4.3) is maintained. As shown in Table 3.1, the dilution time at North Anna for a 100 gpm leak is 44 hours. This is much greater than the frequency of the operator rounds at North Anna, which is once every 6 hours.

3.2.2 Dilution Resulting from the Misalignment of Valves

For North Anna, spent fuel pool dilution events due to the mispositioning of valves is discussed below. There are four sources which interface with the SFP system that could dilute the pool if there is a misalignment of valves:

1. Normal SFP makeup with PG water through the CVCS system blender
2. Refueling Purification system ion exchanger flush connection (referred to as the Demin valve left open in the composite plan analysis)
3. PG lines in the vicinity of the pool
4. Fire Protection emergency fill valve inadvertently opened

3.2.2.1 Makeup Valves Open

As discussed in Section 2.4.1, evaporative losses require that makeup be provided as often as once or twice a week depending on ambient conditions. Normal makeup to the SFP is via one of the two chemical and volume control systems (Unit 1 or 2). Makeup from the CVCS system is through a 2 inch line on the downstream side of the boric acid blender and is isolated by 2 manual valves (1-CH-230 and 1-CH-232). A dilution event could result if both these valves were left open.

The makeup flow may be as much as 200 gpm. Therefore a 200 gpm flowrate is assumed. This results in a dilution time of 22 hours at North Anna (Table 3-1) with over three operator rounds available for detection and response.

3.2.2.2 Demineralizer Valves Open (PG Valves Open)

The PG water system connects directly to the spent fuel pool cooling system at the purification loop demineralizer through a 2 inch line as discussed in Section 2.4.2. The spent fuel pool cooling system is normally isolated by two normally closed manual valves, 1-RP-70 and 1-LW-749. The dilution flowrate through these valves is conservatively estimated to be 200 gpm. Therefore, using a 200 gpm dilution rate, 22 hours (Table 3-1) are available for the operator to detect and respond to the dilution event.

3.2.2.3 PG Valves Open

There are three PG water hose connections in the vicinity of the pool (Section 2.4.4), only one of which is directly next to the pool. The other two point away from the pool and if left open, the water would flow into the pipe trench and not into the pool. Misalignment of the 3/4 inch PG lines around the pool are not analyzed further in this section since it is considered unlikely that an open valve with water flowing onto the floor would go undetected. Additionally, the dilution flow rate (35 gpm) is sufficiently low that it would take numerous operator rounds to not detect it before reaching the criticality limit.

3.2.2.4 Fire Protection Emergency SFP Fill Valve Open

For North Anna, there is an emergency fill line from the underground fire main that may be used in the event of a loss of water in the SFP and a loss of the normal makeup from either unit CVCS system (Section 2.4.5). This line is isolated from the fire main outside the fuel building by a post indicating gate valve (1-FP-56). If the valve was inadvertently opened, 400 gpm of fire water would flow into the SFP. One of the two fire pumps would auto-start, which would annunciate a fire pump start alarm in the control room. The SFP level alarm would also annunciate when the level exceeded its high level setpoint. For a flow rate of 400 gpm, there are 11 hours (Table 3-1) (spans more than one operator round) to detect the dilution event.

3.2.3 Dilution Resulting from Pipe Breaks or Seismic Events

3.2.3.1 Seismic Events

For a seismic event at North Anna, if offsite power is available, the analogous pathway is a rupture of the 3/4" PG water piping, which would cause a leak of approximately 35 gpm. If offsite power is not available, the PG water system would not operate and thus would not be considered

a dilution source. The only other non-seismic piping in the vicinity of the pool is a heating system steam line. If this steam line ruptured, a leak of less than 30 gpm would result, assuming all steam condenses and collects in the SFP. The fire protection piping is seismic, so it is not considered as an additional pathway. Thus, for the seismic event, total dilution flow would be between 30 and 65 gpm depending on whether offsite power is available. As shown in Table 3-1, the dilution time for a 65 gpm leak is 68 hours.

3.2.3.2 Random Pipe Breaks

There are approximately 40 pipe sections in the vicinity of the North Anna spent fuel pool. For North Anna, the safety related CCW and Service Water piping are below the SFP and therefore are not considered to cause a dilution event if they ruptured. The safety related FP piping would flow approximately 350 gpm, if ruptured. From Table 3-1, the time to dilution for a 350 gpm break in the FP line is 13 hours, which spans 2 operator rounds. If the FP pipe ruptured, then one of the two fire pumps would start and an alarm would annunciate in the control room.

For non-safety piping, there are PG lines and plant heating steam lines in the vicinity of the pool, which would flow less than 35 gpm if ruptured. It is assumed, for conservatism, that a plant heating steam line rupture would result in all of the condensed steam collecting in the pool. This is much less than the dilution flows assumed for the composite plant (100 gpm). As a result, with respect to operator detection and response time, approximately 127 hours are available at North Anna for a break in the non-safety piping.

3.2.4 Reactor Cavity Pneumatic Seal Failure

The reactor cavity seal at North Anna contains O-rings instead of inflatable bladders. The O-ring design is less vulnerable to gross failures that could lead to gross loss of inventory. A conservative assumption of a 1000 gpm leak rate is used for assessment of the postulated cavity seal failure. Specifically, if the seal failed, the pool would drain down 25' 3" to the top of the weir between the SFP and transfer canal. Therefore, the pool would drain down in approximately [(25.25/40)* 409,695/1000 gpm] = 258 minutes or 4.3 hours. Thus for North Anna, there are approximately 4 hours available to begin to restore pool inventory before the water is drained to the top of the weir. If a borated water source was not available for makeup, unborated water could be used and boron would then have to be manually added to the pool.

3.2.5 Consideration of Dilution Volumes

As can be seen from Table 3-2, a large volume of water is necessary at North Anna to dilute the spent fuel pool to a condition where $K_{eff} \geq 0.95$. For a dilution event from the nominal spent fuel pool boron concentration of 2300 ppm to a boron endpoint concentration of 1200 ppm, a dilution volume of 266,542 gallons is required. Table 3-2 summarizes the available North Anna dilution sources and their volumes.

TABLE 3-2
North Anna Dilution Sources

Dilution Source	Quantity	Available Water Volume, gal each	Total Water Volume, gal
Primary Grade Reactor Makeup Tanks	2	180,000	360,000
Component Cooling Water	1	3120+[1]	3120+[1]

1 Surge Tank with normal water volume of 3120 gallons, plus system piping.

Although the CCW system has makeup capability from other systems, and the PG Water Tanks contain sufficient dilution volume to exceed the K_{eff} criteria of 0.95, detection of a dilution event via level alarms and/or visual inspections would be expected long before a dilution sufficient to violate the K_{eff} criteria could occur.

4.0 CONCLUSIONS

A boron dilution analysis has been completed for the North Anna spent fuel pool. As a result of this SFP boron dilution analysis, it is concluded that an event which would result in the dilution of the SFP boron concentration from 2300 ppm to 1200 ppm is not a credible event. This conclusion is based on the following:

In order to dilute the SFP to the design K_{eff} of 0.95, a substantial amount of water is needed (266,542 gallons).

Since such a large water volume turnover is required, a SFP dilution event would be readily detected by plant personnel via alarms, flooding or by normal operator rounds through the SFP area.

Evaluations indicate that, based on the flow rates of non-borated water normally available to the SFP, even when significantly higher flowrates are assumed, sufficient time is available to detect and respond to such an event.

It should be noted that this boron dilution evaluation was conducted by evaluating the time and water volumes required to dilute the SFP from 2300 ppm to 1200 ppm. The 1200 ppm end point was utilized to ensure that K_{eff} for the spent fuel racks would remain less than 0.95. As part of the criticality analysis for the North Anna spent fuel pool, a calculation has been performed on a 95/95 basis to show that the spent fuel rack K_{eff} remains less than 1.0 with non-borated water in the pool. Thus, even if the SFP were diluted to zero ppm, which would take significantly more water than evaluated above, the fuel in the racks would be expected to remain subcritical and the health and safety of the public would be protected.

5.0 REFERENCES

1. NRC Correspondence Letter to Mr. Tom Greene, Westinghouse Owners Group Chairman, dated October 25, 1996, titled "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Criticality Analysis Methodology (TAC No. M93254)"

Attachment 2

Mark-up of Unit 1 and Unit 2 Technical Specifications Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

Mark-up of Unit 1 Technical Specifications

~~12-28-79~~

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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NEW SPECIFICATION

3/4.7.14 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 The spent fuel pool boron concentration shall be ≥ 2500 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTION:

a. Immediately suspend movement of fuel assemblies in the spent fuel pool and initiate action to restore the spent fuel pool boron concentration to within limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14 The spent fuel pool boron concentration shall be determined to be ≥ 2500 ppm at least once every 7 days.

New Specification

3/4.7.15 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.7.15 The combination of initial enrichment, burnup, and configuration of the fuel assemblies stored in the spent fuel pool shall be in accordance with the following:

a. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Acceptable" burnup domain of Figure 3.7.15-1 may be stored in the spent fuel pool in a non-matrix location or a low reactivity location in the 5x5 matrix configuration shown in Figure 3.7.15-2. They may also be placed in a high reactivity location if stored in the 5x5 matrix configuration shown in Figure 3.7.15-2.

b. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Conditionally Acceptable" domain of Figure 3.7.15-1 may be stored in the spent fuel pool in a non-matrix location, but must be placed in a high reactivity location if stored in the 5x5 matrix configuration shown in Figure 3.7.15-2

c. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Unacceptable" domain of Figure 3.7.15-1 must be stored in the spent fuel pool in a high reactivity location in the 5x5 matrix configuration shown in Figure 3.7.15-2. A fuel assembly transferred from Surry for storage in the North Anna spent fuel pool must be treated as a fuel assembly in the "Unacceptable" domain.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTION:

a. Immediately initiate action to move the non-complying fuel assembly to an acceptable storage location.

b. The provisions of Specification 3.0.3 are not applicable.

NEW SPECIFICATION

SURVEILLANCE REQUIREMENTS

4.7.15 Prior to storing the fuel assembly in the spent fuel pool location, verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, and storage location of the fuel assembly are in accordance with Specification 3.7.15.

NEW SPECIFICATION

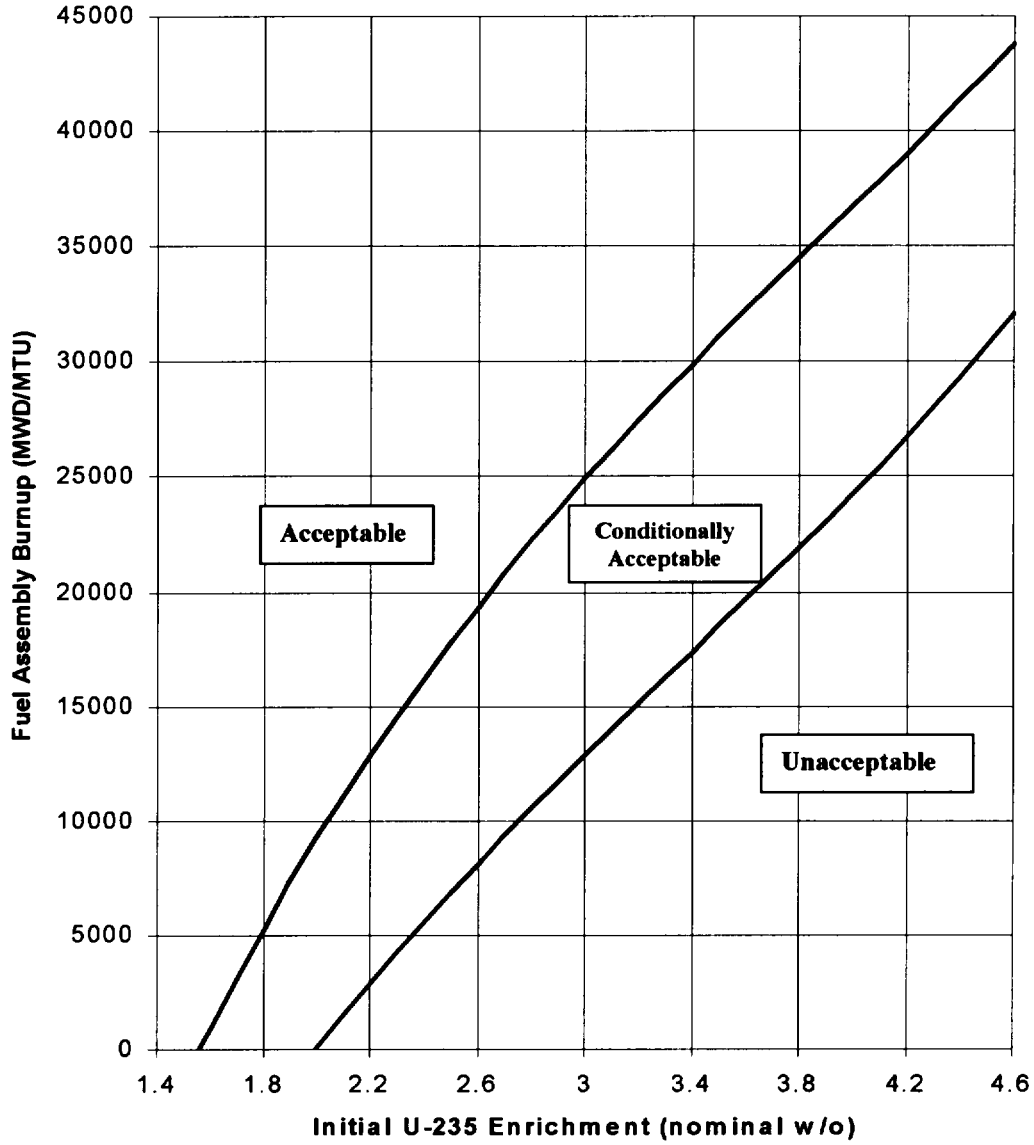


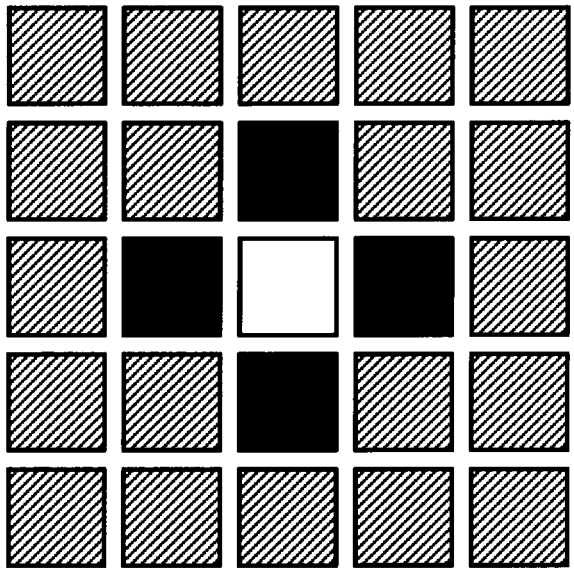
Figure 3.7.15-1: North Anna Burnup Credit Requirements for spent fuel pool storage

Acceptable: Acceptable for storage in non-matrix location or low reactivity location in matrix configuration. May also be placed in high reactivity locations in matrix configuration.

Conditionally Acceptable: Acceptable for storage in non-matrix location, but must be placed in high reactivity location if stored in matrix configuration.

Unacceptable: Must be stored in high reactivity location in matrix configuration. Surry spent fuel must be stored in high reactivity locations in a matrix.

NEW SPECIFICATION



Low reactivity fuel
(Per Figure 3.7.15-1 or cell containing no fuel assembly)

High reactivity fuel
(Per Figure 3.7.15-1, reactivity up to and including 4.6 w/o U^{235} fresh fuel or cell containing no fuel assembly)

No fuel assembly

Figure 3.7.15-2: North Anna 5x5 Matrix Storage Configuration

Notes to Figure:

1. A partial matrix at the boundary of the spent fuel pool storage locations is an acceptable configuration.

2. Storage of non-fueled components within the matrix or non-matrix cells that results in a reduced spent fuel pool K_{eff} is acceptable.

3. A storage cell containing no fuel assembly may be substituted for any location in either matrix or non-matrix configuration.

4. Spent fuel transferred from Surry must be stored in high reactivity locations.

NEW BASIS

BASES

3/4.7.14 SPENT FUEL POOL BORON CONCENTRATION

The specified spent fuel pool boron concentration of ≥ 2500 ppm preserves the assumptions used in the spent fuel pool criticality calculations. The amount of soluble boron required to offset postulated accidents was evaluated for the spent fuel pool. That evaluation established the amount of soluble boron necessary to ensure that K_{eff} will be maintained less than 0.95 should the pool temperature exceed the assumed range or a fuel assembly misload occur. The amount of soluble boron necessary to mitigate these events was determined to be 900 ppm including uncertainties. The specified minimum boron concentration of 2500 ppm assures that the concentration will remain above this value. In addition, the boron concentration of 2500 ppm is consistent with the boron dilution evaluation that demonstrates that any credible dilution event could be terminated prior to reaching the boron concentration for a $K_{eff} \geq 0.95$.

NEW BASIS

BASES

3/4.7.15 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

The restrictions on the placement of fuel assemblies within the spent fuel pool ensure that the K_{eff} will always remain < 0.95, assuming the pool is flooded with borated water. The fuel that may be stored in a non-matrixed configuration is no more reactive than fresh fuel enriched to 2.0 w/o U^{235}. More reactive fuel must be stored in a matrix configuration with less reactive fuel. To ensure that K_{eff} remains < 0.95 for a matrix with the highest reactivity fuel currently allowed by T.S. 5.3.1, the low reactivity fuel in the defined matrix configuration has been defined as equivalent to the reactivity of fresh fuel enriched to ≤ 1.56 w/o U^{235}.

Spent fuel assemblies from Surry Power Station with an initial enrichment of ≤ 4.1 w/o U^{235} are less reactive than North Anna fresh fuel assemblies enriched to 4.6 w/o U^{235}. Therefore, it is acceptable to store Surry spent fuel transferred to North Anna in the high reactivity locations of the 5x5 matrix configuration.

Additional fueled components such as a partially or fully loaded fuel rod basket, a container of fuel pellets/pin fragments (≤ 540 pellets), or fission chamber detectors (< 300 g of U^{235} total for all stored detectors) are less reactive than the low reactivity North Anna spent fuel assemblies. Therefore, these fueled components may be stored in any storage location in the North Anna spent fuel pool where fuel is permitted.

Failed fuel canister storage locations may be filled with fuel assemblies of enrichments < 2.0 w/o U^{235} equivalent and treated as a non-matrix configuration region. Due to the larger pitch of the two failed fuel canister storage locations, the K_{eff} of a non-matrix configuration in the failed fuel assembly storage region is bounded by the K_{eff} of a non-matrix configuration in the nominal spent fuel pool storage region.

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

4.6

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 or ZIRLO. Each fuel rod shall have a nominal active fuel length of 144 inches. 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. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

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 approximately 10,000 cubic feet at nominal operating conditions.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than 1.0 ~~or equal to 0.95~~ when flooded with unborated water, which includes a 2% ~~conservative~~ allowance ~~of 3.4% delta k/k~~ for uncertainties.

b. A nominal 10 9/16 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that, on a best estimate basis, K_{eff} will not exceed .98, with fuel of the highest anticipated enrichment in place, when aqueous foam moderation is assumed.

5.6.1.3 If new fuel for the first core loading is stored dry in the spent fuel storage racks, the center-to-center distance between the new fuel assemblies will be administratively limited to 28 inches and the k_{eff} shall not exceed 0.98 when aqueous foam moderation is assumed.

c. A K_{eff} equivalent to less than 0.95 when fully flooded with water borated to 350 ppm, which includes an allowance for uncertainties but excludes allowances for postulated accidents.

Mark-up of Unit 2 Technical Specifications

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NEW SPECIFICATION

3/4.7.14 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 The spent fuel pool boron concentration shall be ≥ 2500 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTION:

a. Immediately suspend movement of fuel assemblies in the spent fuel pool and initiate action to restore the spent fuel pool boron concentration to within limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14 The spent fuel pool boron concentration shall be determined to be ≥ 2500 ppm at least once every 7 days.

NEW SPECIFICATION

3/4.7.15 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.7.15 The combination of initial enrichment, burnup, and configuration of the fuel assemblies stored in the spent fuel pool shall be in accordance with the following:

a. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Acceptable" burnup domain of Figure 3.7.15-1 may be stored in the spent fuel pool in a non-matrix location or a low reactivity location in the 5x5 matrix configuration shown in Figure 3.7.15-2. They may also be placed in a high reactivity location if stored in the 5x5 matrix configuration shown in Figure 3.7.15-2.

b. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Conditionally Acceptable" domain of Figure 3.7.15-1 may be stored in the spent fuel pool in a non-matrix location, but must be placed in a high reactivity location if stored in the 5x5 matrix configuration shown in Figure 3.7.15-2

c. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Unacceptable" domain of Figure 3.7.15-1 must be stored in the spent fuel pool in a high reactivity location in the 5x5 matrix configuration shown in Figure 3.7.15-2. A fuel assembly transferred from Surry for storage in the North Anna spent fuel pool must be treated as a fuel assembly in the "Unacceptable" domain.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTION:

a. Immediately initiate action to move the non-complying fuel assembly to an acceptable storage location.

b. The provisions of Specification 3.0.3 are not applicable.

NEW SPECIFICATION

SURVEILLANCE REQUIREMENTS

4.7.15 Prior to storing the fuel assembly in the spent fuel pool location, verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, and storage location of the fuel assembly are in accordance with Specification 3.7.15.

NEW SPECIFICATION

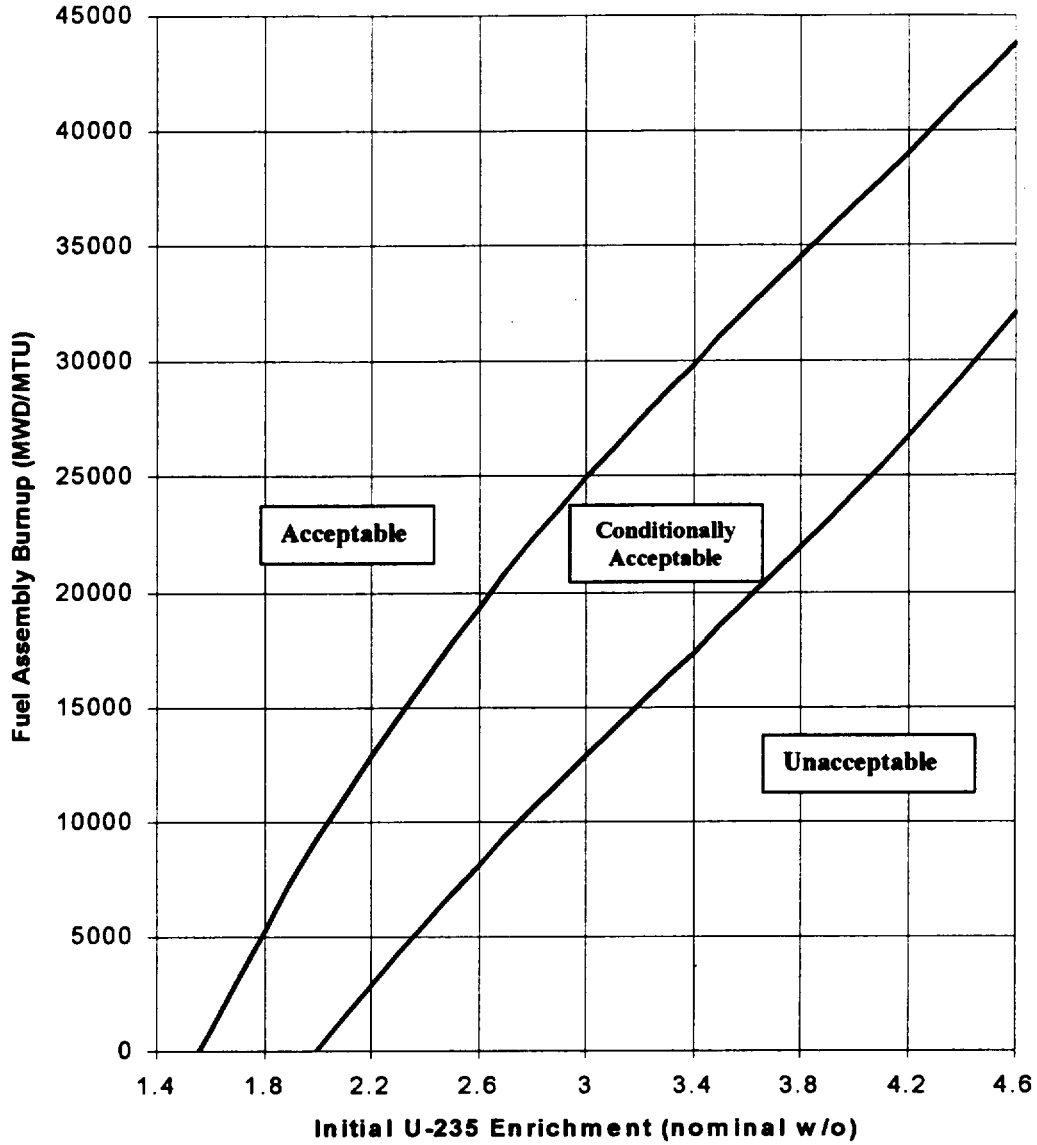


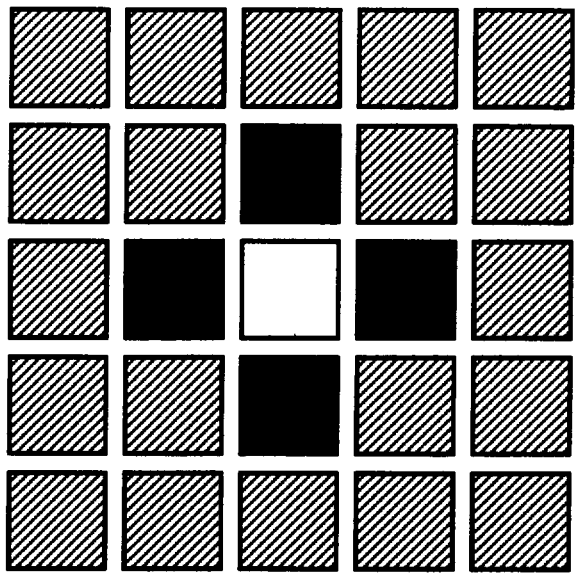
Figure 3.7.15-1: North Anna Burnup Credit Requirements for spent fuel pool storage

Acceptable: Acceptable for storage in non-matrix location or low reactivity location in matrix configuration. May also be placed in high reactivity locations in matrix configuration.

Conditionally Acceptable: Acceptable for storage in non-matrix location, but must be placed in high reactivity location if stored in matrix configuration.

Unacceptable: Must be stored in high reactivity location in matrix configuration. Surry spent fuel must be stored in high reactivity locations in a matrix.

NEW SPECIFICATION



Low reactivity fuel
(Per Figure 3.7.15-1 or cell containing no fuel assembly)

High reactivity fuel
(Per Figure 3.7.15-1, reactivity up to and including 4.6 w/o U^{235} fresh fuel or cell containing no fuel assembly)

No fuel assembly

Figure 3.7.15-2: North Anna 5x5 Matrix Storage Configuration

Notes to Figure:

1. A partial matrix at the boundary of the spent fuel pool storage locations is an acceptable configuration.

2. Storage of non-fueled components within the matrix or non-matrix cells that results in a reduced spent fuel pool K_{eff} is acceptable.

3. A storage cell containing no fuel assembly may be substituted for any location in either matrix or non-matrix configuration.

4. Spent fuel transferred from Surry must be stored in high reactivity locations.

NEW BASIS

BASES

3/4.7.14 SPENT FUEL POOL BORON CONCENTRATION

The specified spent fuel pool boron concentration of ≥ 2500 ppm preserves the assumptions used in the spent fuel pool criticality calculations. The amount of soluble boron required to offset postulated accidents was evaluated for the spent fuel pool. That evaluation established the amount of soluble boron necessary to ensure that K_{eff} will be maintained less than 0.95 should the pool temperature exceed the assumed range or a fuel assembly misload occur. The amount of soluble boron necessary to mitigate these events was determined to be 900 ppm including uncertainties. The specified minimum boron concentration of 2500 ppm assures that the concentration will remain above this value. In addition, the boron concentration of 2500 ppm is consistent with the boron dilution evaluation that demonstrates that any credible dilution event could be terminated prior to reaching the boron concentration for a $K_{eff} \geq 0.95$.

NEW BASIS

BASES

3/4.7.15 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

The restrictions on the placement of fuel assemblies within the spent fuel pool ensure that the K_{eff} will always remain < 0.95, assuming the pool is flooded with borated water. The fuel that may be stored in a non-matrixed configuration is no more reactive than fresh fuel enriched to 2.0 w/o U^{235}. More reactive fuel must be stored in a matrix configuration with less reactive fuel. To ensure that K_{eff} remains < 0.95 for a matrix with the highest reactivity fuel currently allowed by T.S. 5.3.1, the low reactivity fuel in the defined matrix configuration has been defined as equivalent to the reactivity of fresh fuel enriched to ≤ 1.56 w/o U^{235}.

Spent fuel assemblies from Surry Power Station with an initial enrichment of ≤ 4.1 w/o U^{235} are less reactive than North Anna fresh fuel assemblies enriched to 4.6 w/o U^{235}. Therefore, it is acceptable to store Surry spent fuel transferred to North Anna in the high reactivity locations of the 5x5 matrix configuration.

Additional fueled components such as a partially or fully loaded fuel rod basket, a container of fuel pellets/pin fragments (≤ 540 pellets), or fission chamber detectors (< 300 g of U^{235} total for all stored detectors) are less reactive than the low reactivity North Anna spent fuel assemblies. Therefore, these fueled components may be stored in any storage location in the North Anna spent fuel pool where fuel is permitted.

Failed fuel canister storage locations may be filled with fuel assemblies of enrichments < 2.0 w/o U^{235} equivalent and treated as a non-matrix configuration region. Due to the larger pitch of the two failed fuel canister storage locations, the K_{eff} of a non-matrix configuration in the failed fuel assembly storage region is bounded by the K_{eff} of a non-matrix configuration in the nominal spent fuel pool storage region.

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 or ZIRLO. Each fuel rod shall have a nominal active fuel length of 144 inches. 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~~ 4.6 weight percent U-235. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

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 approximately 10,000 cubic feet at nominal operating conditions.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than 1.0 ~~or equal to 0.95~~ when flooded with unborated water, which includes an ~~a conservative~~ allowance ~~of 3.4% delta k/k~~ for uncertainties.

b. A nominal 10 9/16 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that, on a best estimate basis, k_{eff} will not exceed .98, with fuel of the highest anticipated enrichment in place, when aqueous foam moderation is assumed.

5.6.1.3 If new fuel for the first core loading is stored dry in the spent fuel storage racks, the center-to-center distance between the new fuel assemblies will be administratively limited to 28 inches and the k_{eff} shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 288.83 feet Mean Sea Level, USGS datum.

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1737 fuel assemblies.

c. A K_{eff} equivalent to less than 0.95 when fully flooded with water borated to 350 ppm, which includes an allowance for uncertainties but excludes allowances for postulated accidents.

Attachment 3

Proposed Unit 1 and Unit 2 Technical Specifications Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

TECH SPEC CHANGE REQUEST NO. 339

TABULATION OF CHANGES

License No. NPF-4 / Docket No. 50-338

Summary of change:

This proposed change to the Technical Specifications is being made to 1) increase the fuel enrichment limit to 4.6 weight percent Uranium^{235}, 2) establish Technical Specifications Limiting Conditions for Operations for Spent Fuel Pool (SFP) boron concentration and fuel storage restrictions, and 3) eliminate the value for the allowance for uncertainties in the calculation for Keff in the SFP criticality calculation.

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Proposed Technical Specifications Changes

for North Anna Unit 1

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3/4.7.14 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 The spent fuel pool boron concentration shall be ≥ 2500 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTION:

a. Immediately suspend movement of fuel assemblies in the spent fuel pool and initiate action to restore the spent fuel pool boron concentration to within limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14 The spent fuel pool boron concentration shall be determined to be ≥ 2500 ppm at least once every 7 days.

3/4.7.15 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.7.15 The combination of initial enrichment, burnup, and configuration of the fuel assemblies stored in the spent fuel pool shall be in accordance with the following:

a. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Acceptable" burnup domain of Figure 3.7.15-1 may be stored in the spent fuel pool in a non-matrix location or a low reactivity location in the 5 x 5 matrix configuration shown in Figure 3.7.15-2. They may also be placed in a high reactivity location if stored in the 5 x 5 matrix configuration shown in Figure 3.7.15-2.

b. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Conditionally Acceptable" domain of Figure 3.7.15-1 may be stored in the spent fuel pool in a non-matrix location, but must be placed in a high reactivity location if stored in the 5 x 5 matrix configuration shown in Figure 3.7.15-2.

c. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Unacceptable" domain of Figure 3.7.15-1 must be stored in the spent fuel pool in a high reactivity location in the 5 x 5 matrix configuration shown in Figure 3.7.15-2. A fuel assembly transferred from Surry for storage in the North Anna spent fuel pool must be treated as a fuel assembly in the "Unacceptable" domain.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTION:

a. Immediately initiate action to move the non-complying fuel assembly to an acceptable storage location.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.15 Prior to storing the fuel assembly in the spent fuel pool location, verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, and storage location of the fuel assembly are in accordance with Specification 3.7.15.

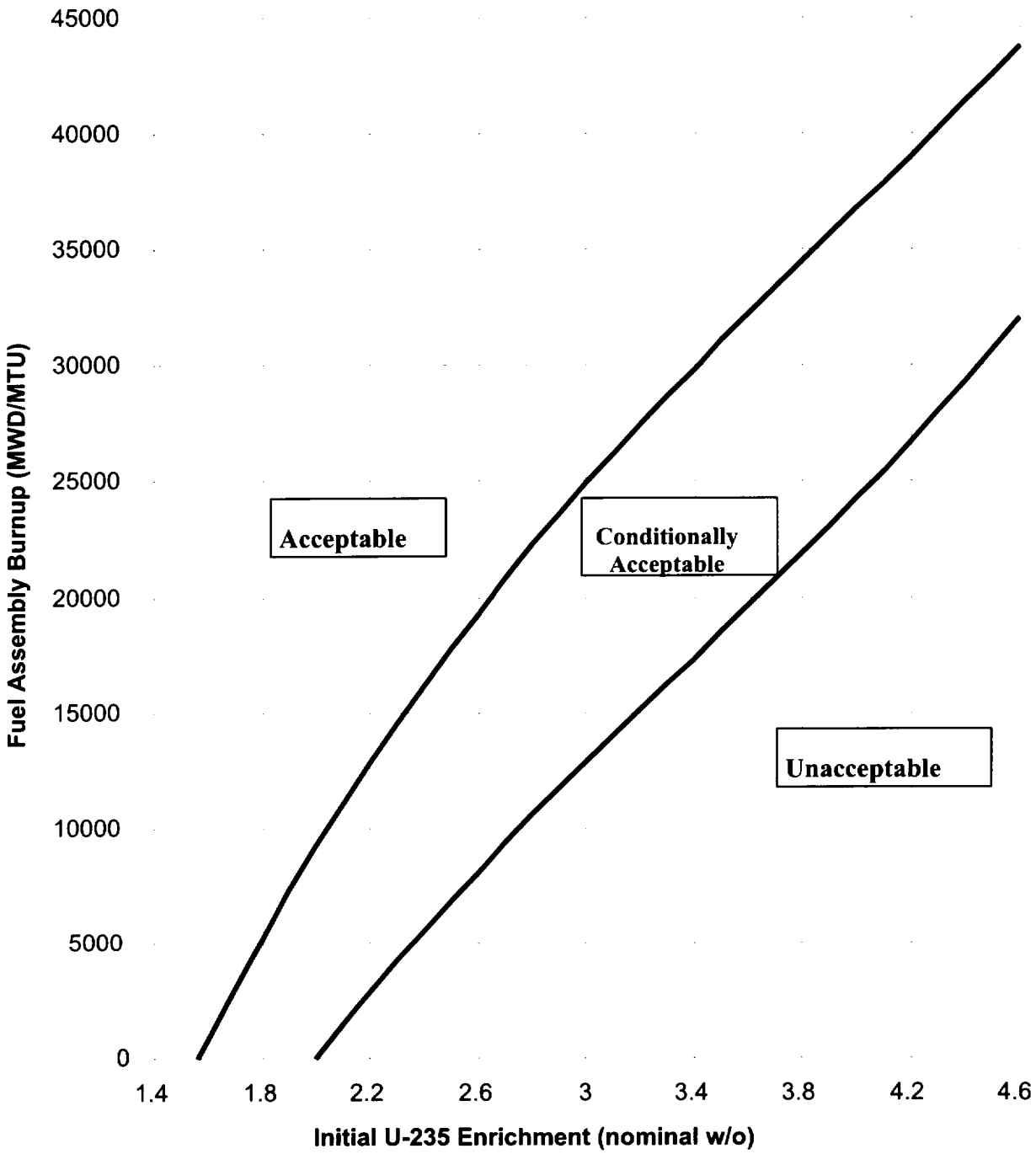
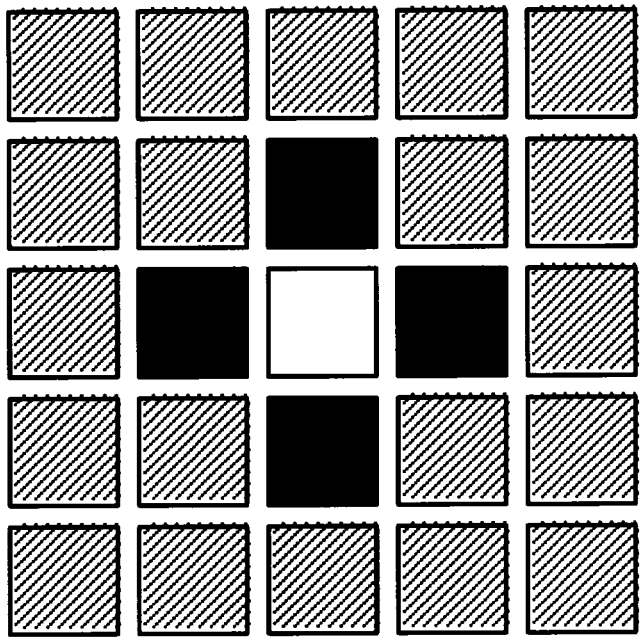


Figure 3.7.15-1 North Anna Burnup Credit Requirements for Spent Fuel Pool Storage

Acceptable: Acceptable for storage in non-matrix location or low reactivity location in matrix configuration. May also be placed in high reactivity locations in matrix configuration.

Conditionally Acceptable: Acceptable for storage in non-matrix location, but must be placed in high reactivity location if stored in matrix configuration.

Unacceptable: Must be stored in high reactivity location in matrix configuration. Surry spent fuel must be stored in high reactivity locations in a matrix.



Low reactivity fuel
(Per Figure 3.7.15-1 or cell containing no fuel assembly)

High reactivity fuel
(Per Figure 3.7.15-1, reactivity up to and including 4.6 w/o U^{235} fresh fuel or cell containing no fuel assembly)

No fuel assembly

Figure 3.7.15-2 North Anna 5 x 5 Matrix Storage Configuration

Notes to Figure:

1. A partial matrix at the boundary of the spent fuel pool storage locations is an acceptable configuration.
2. Storage of non-fueled components within the matrix or non-matrix cells that results in a reduced spent fuel pool K_{eff} is acceptable.
3. A storage cell containing no fuel assembly may be substituted for any location in either matrix or non-matrix configuration.
4. Spent fuel transferred from Surry must be stored in high reactivity locations.

3/4.7.14 SPENT FUEL POOL BORON CONCENTRATION

The specified spent fuel pool boron concentration of ≥ 2500 ppm preserves the assumptions used in the spent fuel pool criticality calculations. The amount of soluble boron required to offset postulated accidents was evaluated for the spent fuel pool. That evaluation established the amount of soluble boron necessary to ensure that K_{eff} will be maintained less than 0.95 should the pool temperature exceed the assumed range or a fuel assembly misload occur. The amount of soluble boron necessary to mitigate these events was determined to be 900 ppm including uncertainties. The specified minimum boron concentration of 2500 ppm assures that the concentration will remain above this value. In addition, the boron concentration of 2500 ppm is consistent with the boron dilution evaluation that demonstrates that any credible dilution event could be terminated prior to reaching the boron concentration for a $K_{eff} \geq 0.95$.

3/4.7.15 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

The restrictions on the placement of fuel assemblies within the spent fuel pool ensure that the K_{eff} will always remain < 0.95, assuming the pool is flooded with borated water. The fuel that may be stored in a non-matrixed configuration is no more reactive than fresh fuel enriched to 2.0 w/o U^{235}. More reactive fuel must be stored in a matrix configuration with less reactive fuel. To ensure that K_{eff} remains < 0.95 for a matrix with the highest reactivity fuel currently allowed by T.S. 5.3.1, the low reactivity fuel in the defined matrix configuration has been defined as equivalent to the reactivity of fresh fuel enriched to ≤ 1.56 w/o U^{235}.

Spent fuel assemblies from Surry Power Station with an initial enrichment of ≤ 4.1 w/o U^{235} are less reactive than North Anna fresh fuel assemblies enriched to 4.6 w/o U^{235}. Therefore, it is acceptable to store Surry spent fuel transferred to North Anna in the high reactivity locations of the 5 x 5 matrix configuration.

Additional fueled components such as a partially or fully loaded fuel rod basket, a container of fuel pellets/pin fragments (≤ 540 pellets), or fission chamber detectors (< 300 g of U^{235} total for all stored detectors) are less reactive than the low reactivity North Anna spent fuel assemblies. Therefore, these fueled components may be stored in any storage location in the North Anna spent fuel pool where fuel is permitted.

Failed fuel canister storage locations may be filled with fuel assemblies of enrichments < 2.0 w/o U^{235} equivalent and treated as a non-matrix configuration region. Due to the larger pitch of the two failed fuel canister storage locations, the K_{eff} of a non-matrix configuration in the failed fuel assembly storage region is bounded by the K_{eff} of a non-matrix configuration in the nominal spent fuel pool storage region.

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 or ZIRLO. Each fuel rod shall have a nominal active fuel length of 144 inches. 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 4.6 weight percent U-235. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

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 approximately 10,000 cubic feet at nominal operating conditions.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than 1.0 when flooded with unborated water, which includes an allowance for uncertainties.

b. A nominal 10 9/16 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. A K_{eff} equivalent to less than 0.95 when fully flooded with water borated to 350 ppm, which includes an allowance for uncertainties but excludes allowances for postulated accidents.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that, on a best estimate basis, K_{eff} will not exceed .98, with fuel of the highest anticipated enrichment in place, when aqueous foam moderation is assumed.

5.6.1.3 If new fuel for the first core loading is stored dry in the spent fuel storage racks, the center-to-center distance between the new fuel assemblies will be administratively limited to 28 inches and the k_{eff} shall not exceed 0.98 when aqueous foam moderation is assumed.

TECH SPEC CHANGE REQUEST NO. 379

TABULATION OF CHANGES

License No. NPF-7 / Docket No. 50-339

Summary of change:

This proposed change to the Technical Specifications is being made to 1) increase the fuel enrichment limit to 4.6 weight percent Uranium^{235}, 2) establish Technical Specifications Limiting Conditions for Operations for Spent Fuel Pool (SFP) boron concentration and fuel storage restrictions, and 3) eliminate the value for the allowance for uncertainties in the calculation for Keff in the SFP criticality calculation.

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Proposed Technical Specifications Changes

for North Anna Unit 2

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3/4.7.14 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 The spent fuel pool boron concentration shall be ≥ 2500 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool

ACTION:

a. Immediately suspend movement of fuel assemblies in the spent fuel pool and initiate action to restore the spent fuel pool boron concentration to within limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14 The spent fuel pool boron concentration shall be determined to be ≥ 2500 ppm at least once every 7 days.

3/4.7.15 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.15 The combination of initial enrichment, burnup, and configuration of the fuel assemblies stored in the spent fuel pool shall be in accordance with the following:

a. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Acceptable" burnup domain of Figure 3.7.15-1 may be stored in the spent fuel pool in a non-matrix location or a low reactivity location in the 5 x 5 matrix configuration shown in Figure 3.7.15-2. They may also be placed in a high reactivity location if stored in the 5 x 5 matrix configuration shown in Figure 3.7.15-2.

b. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Conditionally Acceptable" domain of Figure 3.7.15-1 may be stored in the spent fuel pool in a non-matrix location, but must be placed in a high reactivity location if stored in the 5 x 5 matrix configuration shown in Figure 3.7.15-2.

c. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "Unacceptable" domain of Figure 3.7.15-1 must be stored in the spent fuel pool in a high reactivity location in the 5 x 5 matrix configuration shown in Figure 3.7.15-2. A fuel assembly transferred from Surry for storage in the North Anna spent fuel pool must be treated as a fuel assembly in the "Unacceptable" domain.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTION:

a. Immediately initiate action to move the non-complying fuel assembly to an acceptable storage location.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.15 Prior to storing the fuel assembly in the spent fuel pool location, verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, and storage location of the fuel assembly are in accordance with Specification 3.7.15.

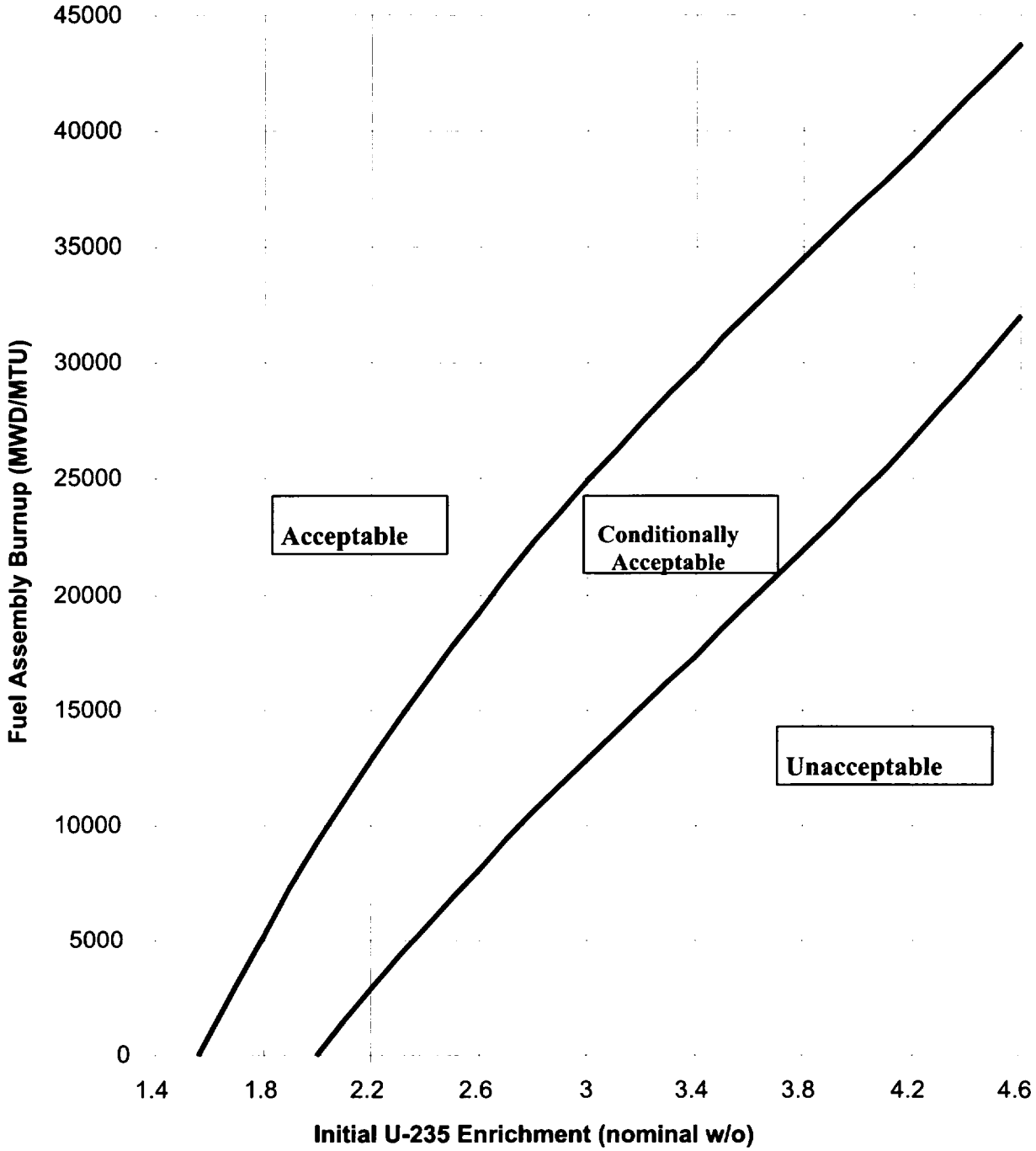
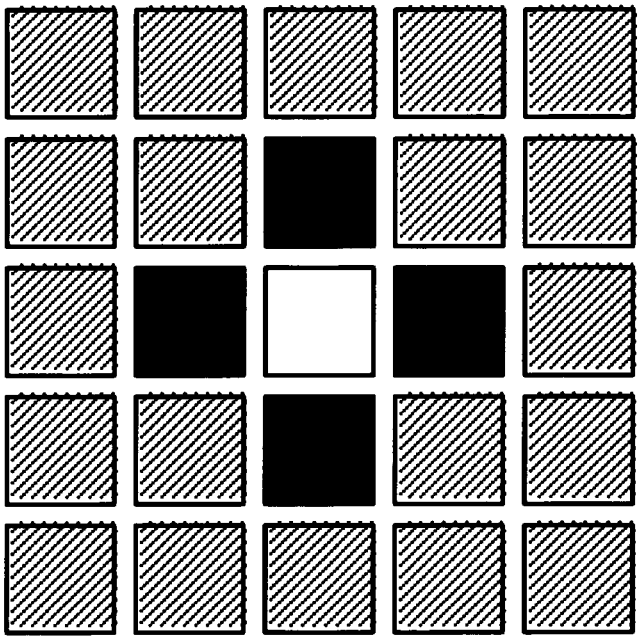


Figure 3.7.15-1 North Anna Burnup Credit Requirements for Spent Fuel Pool Storage

Acceptable: Acceptable for storage in non-matrix location or low reactivity location in matrix configuration. May also be placed in high reactivity locations in matrix configuration.

Conditionally Acceptable: Acceptable for storage in non-matrix location, but must be placed in high reactivity location if stored in matrix configuration.

Unacceptable: Must be stored in high reactivity location in matrix configuration. Surry spent fuel must be stored in high reactivity locations in a matrix.



Low reactivity fuel
(Per Figure 3.7.15-1 or cell containing no fuel assembly)

High reactivity fuel
(Per Figure 3.7.15-1, reactivity up to and including 4.6 w/o U^{235} fresh fuel or cell containing no fuel assembly)

No fuel assembly

Figure 3.7.15-2 North Anna 5 x 5 Matrix Storage Configuration

Notes to Figure:

1. A partial matrix at the boundary of the spent fuel pool storage locations is an acceptable configuration.
2. Storage of non-fueled components within the matrix or non-matrix cells that results in a reduced spent fuel pool K_{eff} is acceptable.
3. A storage cell containing no fuel assembly may be substituted for any location in either matrix or non-matrix configuration.
4. Spent fuel transferred from Surry must be stored in high reactivity locations.

3/4.7.14 SPENT FUEL POOL BORON CONCENTRATION

The specified spent fuel pool boron concentration of ≥ 2500 ppm preserves the assumptions used in the spent fuel pool criticality calculations. The amount of soluble boron required to offset postulated accidents was evaluated for the spent fuel pool. That evaluation established the amount of soluble boron necessary to ensure that K_{eff} will be maintained less than 0.95 should the pool temperature exceed the assumed range or a fuel assembly misload occur. The amount of soluble boron necessary to mitigate these events was determined to be 900 ppm including uncertainties. The specified minimum boron concentration of 2500 ppm assures that the concentration will remain above this value. In addition, the boron concentration of 2500 ppm is consistent with the boron dilution evaluation that demonstrates that any credible dilution event could be terminated prior to reaching the boron concentration for a $K_{eff} \geq 0.95$.

3/4.7.15 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

The restrictions on the placement of fuel assemblies within the spent fuel pool ensure that the K_{eff} will always remain < 0.95, assuming the pool is flooded with borated water. The fuel that may be stored in a non-matrixed configuration is no more reactive than fresh fuel enriched to 2.0 w/o U^{235}. More reactive fuel must be stored in a matrix configuration with less reactive fuel. To ensure that K_{eff} remains < 0.95 for a matrix with the highest reactivity fuel currently allowed by T.S. 5.3.1, the low reactivity fuel in the defined matrix configuration has been defined as equivalent to the reactivity of fresh fuel enriched to ≤ 1.56 w/o U^{235}.

Spent fuel assemblies from Surry Power Station with an initial enrichment of ≤ 4.1 w/o U^{235} are less reactive than North Anna fresh fuel assemblies enriched to 4.6 w/o U^{235}. Therefore, it is acceptable to store Surry spent fuel transferred to North Anna in the high reactivity locations of the 5 x 5 matrix configuration.

Additional fueled components such as a partially or fully loaded fuel rod basket, a container of fuel pellets/pin fragments (≤ 540 pellets), or fission chamber detectors (< 300 g of U^{235} total for all stored detectors) are less reactive than the low reactivity North Anna spent fuel assemblies. Therefore, these fueled components may be stored in any storage location in the North Anna spent fuel pool where fuel is permitted.

Failed fuel canister storage locations may be filled with fuel assemblies of enrichments < 2.0 w/o U^{235} equivalent and treated as a non-matrix configuration region. Due to the larger pitch of the two failed fuel canister storage locations, the K_{eff} of a non-matrix configuration in the failed fuel assembly storage region is bounded by the K_{eff} of a non-matrix configuration in the nominal spent fuel pool storage region.

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 or ZIRLO. Each fuel rod shall have a nominal active fuel length of 144 inches. 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 4.6 weight percent U-235. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

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 approximately 10,000 cubic feet at nominal operating conditions.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than 1.0 when flooded with unborated water, which includes an allowance for uncertainties.

b. A nominal 10 9/16 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. A K_{eff} equivalent to less than 0.95 when fully flooded with water borated to 350 ppm, which includes an allowance for uncertainties but excludes allowances for postulated accidents.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that, on a best estimate basis, k_{eff} will not exceed .98, with fuel of the highest anticipated enrichment in place, when aqueous foam moderation is assumed.

5.6.1.3 If new fuel for the first core loading is stored dry in the spent fuel storage racks, the center-to-center distance between the new fuel assemblies will be administratively limited to 28 inches and the k_{eff} shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 288.83 feet Mean Sea Level, USGS datum.

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1737 fuel assemblies.

Attachment 4

Significant Hazards Consideration Determination

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Virginia Power) proposes to establish Technical Specifications requirements for spent fuel pool (SFP) soluble boron concentration and fuel assembly loading restrictions based on burnup and enrichment. These new Technical Specifications requirements will permit elimination of the Boraflex credit from the spent fuel pool criticality calculations. In addition, we propose increasing the maximum fuel enrichment for North Anna Units 1 and 2 from the current Technical Specifications limit of 4.3 weight percent U^{235} to 4.6 weight percent U^{235}. This higher enrichment has been incorporated into the new criticality calculations for the spent fuel pool and fresh fuel storage racks.

The increase in enrichment, in conjunction with the previously submitted Technical Specifications change to increase the soluble boron concentration in plant safety systems submitted June 22, 2000, will reduce the need for extended periods of reduced power operation (coastdowns) at the end of each operating cycle and permit fuel discharge burnups more compatible with the current lead rod limit of 60,000 MWD/MTU. This will help optimize fuel cycle costs while continuing to satisfy the current core power distribution and safety limits.

The environmental effects of the higher enrichment fuel were reviewed, as were impacts to the North Anna safety analyses. It was determined that the generic fuel data (temperature and pressures) provided by Westinghouse for input to safety analyses are valid for fuel enrichments which exceed the requested change, so inputs to existing safety analyses are unaffected by the proposed enrichment increase. The consequences of accident scenarios are also unchanged, because the source terms used to determine the releases from fuel during accidents are a function of burnup, rather than initial enrichment. As the operating power and fuel burnup limits are not being changed when the fuel enrichment is increased, it is concluded that there are no adverse effects on the types or amounts of any radiological releases.

Other areas identified as potentially being affected by the enrichment increase included the impact on reactor vessel fluence, control rod insertion, decay heat load in the SFP during a full core off load, dry fuel storage, core physics parameters and rod integrity. The higher enrichment and change to the SFP design bases has no appreciable effect on these areas except for the core parameters. Of the core parameters, existing limits for peaking factors, MTC, FTC and control rod worths will continue to be met. However, an increase in the critical boron concentration is expected. The critical boron concentration is evaluated as part of the reload design process, so the impact of any required increase will be calculated prior to each reload to ensure compliance with all applicable Technical Specifications.

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

1. The proposed increase in maximum fuel enrichment or the changes to the SFP design basis will not increase the probability of an accident previously evaluated in the North Anna Units 1 and 2 UFSAR. The only accidents for which the probability of occurrence is potentially affected by the fuel enrichment and SFP changes involve criticality events during fuel handling and storage (e.g., fuel mispositioning). Positioning of fuel assemblies in the SFP has always been administratively controlled. The new Technical Specifications place additional restrictions on the determination of the appropriate position in the SFP for each fuel assembly, but these considerations are incorporated into our administrative tools and guidelines. The controls on fuel movement, including any checks to ensure the assembly is placed in the specified location, remain unchanged. Criticality safety analyses have been performed that demonstrate that the K_{eff} during the handling and storage of both new and spent fuel is low enough to ensure subcriticality during postulated accident conditions. In addition safety analyses of the dilution of the spent fuel pool have been performed to ensure that there is adequate time for a dilution accident to be found and mitigated before criticality is reached in the spent fuel pool. 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 limits already approved, radiological consequences of other accidents previously evaluated in the North Anna Units 1 and 2 UFSAR will not be increased.

2. The possibility of an accident which is different from any already discussed in the North Anna Units 1 and 2 UFSAR is not created. Although there are new restrictions on placement of fuel in the SFP, the controls on fuel movement to the administratively specified locations in the pool are unchanged. The higher enrichment fuel and the new Technical Specifications for the spent fuel pool do not require any new or different plant equipment, and do not change the manner in which currently installed equipment is operated. There are no changes to normal core operation, and the units 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. Existing safety analyses of record will remain applicable for use of fuel with the higher initial enrichment.

3. The probability of a malfunction of equipment important to safety previously evaluated in the North Anna Units 1 and 2 UFSAR is not increased. The design of cores which incorporate fuel at the higher initial enrichment and the spent fuel pool 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, including fuel mispositioning and pool dilution scenarios. The new Technical Specifications on the spent fuel pool do not introduce any new effluents or release paths, and do not affect the magnitude of any currently analyzed releases. Also, since the burnup limit of the fuel is not being increased, the radiological consequences of any malfunction of equipment important to safety previously evaluated in the North Anna UFSAR is not increased by the use of a higher fuel enrichment limit.

4. The possibility of a malfunction of equipment important to safety different from any already evaluated in the North Anna Units 1 and 2 UFSAR is not created. The design for North Anna cycles, which incorporate the higher enriched fuel, will meet all applicable design criteria. Fuel handling and storage in the spent fuel pool with additional restrictions on the selection of fuel storage locations and increased boron concentration will also 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. No new or different plant equipment is introduced, and the operation of currently installed equipment is not changed. The use of higher enriched fuel and the changes to the SFP design basis have the potential to affect only criticality events during fuel handling and storage. Safety analyses demonstrated that K_{eff} will remain sufficiently low to ensure subcriticality, so no new releases will result and there is no impact on radiological consequences of accidents.

5. The margin of safety as defined in the Bases to any North Anna Technical Specification is not reduced. Safety analyses of record will remain applicable for the operation of fuel with a higher initial U^{235} enrichment and changes to the spent fuel pool. Criticality analyses demonstrate that the limits on K_{eff} 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 North Anna Units 1 and 2 Technical Specifications 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 North Anna 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 or changes to the spent fuel pool design basis.

Based on the evaluations and analyses results presented in the foregoing safety significance evaluation, it has been demonstrated that increasing the North Anna Units 1 and 2 maximum initial fuel enrichment to 4.6 weight percent U^{235} and changing the design basis of the spent fuel pool to eliminate any credit for Boraflex but take credit for soluble boron in the pool 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.