June 15, 2001

Mr. David A. Christian
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATIONS CHANGES TO INCREASE FUEL ENRICHMENT AND SPENT FUEL POOL SOLUBLE BORON AND FUEL BURNUP CREDIT (TAC NOS. MB0197 AND MB0198)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment Nos. 227 and 208 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2. The amendments change the Technical Specifications (TS) in response to your letter dated September 27, 2000, as supplemented November 21 and December 18, 2000, and February 2, March 2, and May 21, 2001.

These amendments add Technical Specification (TS) 3.7.14, TS 4.7.14, TS 3.7.15, TS 4.7.15, Figure 3.7.15-1, and Figure 3.7.15-2; and revise TS 5.3.1 and TS 5.6.1.1. The purpose of these amendments is to increase the limit on the fuel enrichment from the current limit of 4.3 weight percent U^{235} to a maximum of 4.6 weight percent U^{235}, establish TS Limiting Conditions for Operations for the Spent Fuel Pool (SFP) boron concentration and fuel storage restrictions, and eliminate the value of uncertainties in the calculation for Keff in the SFP criticality calculation.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Stephen Monarque, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:

1. Amendment No. 227 to NPF-4
2. Amendment No. 208 to NPF-7
3. Safety Evaluation

cc w/encls: See next page

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNITS NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 227
License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated September 27, 2000, as supplemented November 21 and December 18, 2000, and February 2, March 2, and May 21, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 227 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented by December 31, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments:
Changes to the Technical
Specifications

Date of Issuance: June 15, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 227

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
VIII	VIII
XV	XV
------	3/4 7-75
------	3/4 7-76
------	3/4 7-77
------	3/4 7-78
B 3/4 7-10	B 3/4 7-10
5-4	5-4
5-5	5-5

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 208
License No. NPF-7

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated September 27, 2000, as supplemented November 21, and December 18, 2000, and February 2, March 2, and May 21, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 208 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented by December 31, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments:
Changes to the Technical
Specifications

Date of Issuance: June 15, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 208

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
VII	VII
XIII	XIII
------	3/4 7-59
------	3/4 7-60
------	3/4 7-61
------	3/4 7-62
B 3/4 7-10	B 3/4 7-10
5-4	5-4
5-5	5-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 227 TO FACILITY OPERATING LICENSE NO. NPF-4

AND AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated September 27, 2000, as supplemented November 21 and December 18, 2000, and February 2, March 2, and May 21, 2001, Virginia Electric and Power Company, (the licensee) requested changes to the Technical Specifications (TS) for the Facility Operating Licenses for North Anna Power Station (NAPS), Units 1 and 2. The November 21, 2000, submittal provided a No Significant Hazards Consideration Determination (NSHCD) statement that superseded the NSHCD in the September 27, 2000, submittal. The proposed changes are:

a. Increase the limit on the feed fuel enrichment from the current maximum of 4.3 wt % U^{235} to a maximum of 4.6 wt % U^{235}.

b. Establish TS Limiting Conditions for Operation (LCO) with regard to Spent Fuel Pool (SFP) boron concentration and establish administrative restrictions on fuel storage location based on burnup and enrichment to accommodate the increase in the maximum fuel enrichment.

c. Eliminate the value for the allowance for uncertainties in the calculation for k_{eff} in the SFP criticality calculation.

The purpose of increasing the fuel enrichment limit is to better optimize the fuel cycle costs by reducing the need for operation at lower power at the end of each operating cycle under the current fuel management of 18-month cycles at 98% load factor.

2.0 EVALUATION

2.1 Enrichment Limit Increase and Core Physics Considerations

The proposed TS change to the limit on the feed fuel enrichment from 4.3 wt % U^{235} to 4.6 wt % U^{235} affects the core physics parameters directly, since the current fuel management strategy of a very low leakage core, which is based on a checkerboard pattern with fresh fuel preferentially put in the interior of the core, will be maintained during the transition and equilibrium cores.

The licensee's estimates of the impact of the enrichment change on physics parameters based on an equilibrium core with a batch loading of 28 assemblies at 4.5 wt % U^{235} and 36 assemblies at the 4.6 wt % U^{235} limit are appropriate. The licensee has submitted estimates of the expected changes in core physics parameters, such as peaking factors, reactivity coefficients, boron concentrations, and rod worths due to the proposed increase of the enrichment limit. These estimated values are reasonable, and there is reasonable assurance that safety limits will not be exceeded (Ref. 1). Moreover, an assessment of nuclear-related key safety parameters is made during operation before each reload (Ref. 2). In addition, explicit account of the effects of the physics parameters is taken in the predictions for certification of surveillance limits, startup physics testing, and for physics parameter operator curves that are used as input to station operating procedures. This, in conjunction with the fact that there are no changes to any of the NAPS TS LCO, operating or safety-related setpoints, or TS bases associated with this enrichment increase, makes the proposed TS change to the fuel enrichment limit acceptable in the context of its impact on core physics parameters.

2.2 Fuel Storage

The issue with regard to fuel storage when the feed fuel enrichment is increased is whether or not the configuration of fuel assemblies continues to be subcritical with sufficient margin. To address this issue, General Design Criterion (GDC)-62 "Prevention of Criticality in Fuel Storage and Handling" (Ref. 3) specifies that "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

No physical modifications are to be made to the current NAPS fuel storage racks with regard to geometrical configuration in order to accommodate the impact of the more highly enriched fuel. Rather, it is proposed by the licensee that partial credit be taken for the physical processes associated with the SFP boron concentration, and the administrative restrictions put in place on fuel storage location based on the decrease of fissile material due to burnup and the difference in the initial feed enrichment between assemblies. The administrative controls consist of three basic components: (1) the generation of accurate fuel assembly enrichment and burnup data; (2) the generation of a fuel handling report prior to and for each fuel movement, based on the previously computed information; and (3) procedures that not only govern the preparation of the fuel handling report, but ensure that the fuel movement is carried out in accordance with the fuel handling report. The staff's evaluation of these proposed administrative controls and the use of soluble boron to prevent criticality in the SFP is set forth below.

2.2.1 New Fuel Storage Area

For the fresh fuel storage area, at issue is the assurance of the subcriticality of the SFP under normal and postulated abnormal rack conditions, while loaded with fresh fuel enriched to 4.6 wt % U^{235}. In particular, NRC guidelines state that k_{eff} should not equal or exceed 0.95 under the assumption that the fresh fuel storage rack is dry or flooded with unborated water, and should not exceed 0.98 under optimum moderation. The limiting postulated accident is determined as a flooding accident and is taken into account in the criticality analysis by assuming the nominal k_{eff} to be at optimum moderation (i.e., at its maximum value in this context).

The computed estimate of the maximum k_{eff}, in the case of fresh fuel, has three components which take into account biases and uncertainties: (1) a nominal value at optimum moderation; (2) a calculational methods bias (i.e., the difference between the computed and measured k_{eff} of a critical assembly); and (3) a bias which takes into account variation in rack geometric and fuel composition and design dimensions due to manufacturing tolerances. The basis for the methodology for this computation, and applied by the licensee, has been previously reviewed and approved by the NRC staff (Refs. 4 and 5).

The k_{eff} values are computed using a 3D KENO-V model and ENDF/B-V 238-group cross sections. These are standard tools for criticality analysis (Ref. 6). In order to verify the computational model and its implementation, the licensee has performed calculations of 59 relevant critical experiments (Ref. 7). The results are consistent with those reported in Ref. 8. These calculations also give the estimate of the methods bias and its uncertainty. The licensee applies these results to the computation of the methods bias by taking the difference between average measured k_{eff} and the computed k_{eff} for the critical experiments. The uncertainty is computed as the standard deviation in the differences of the measured and computed K_{eff} values.

The bias and uncertainty for taking account of dimensional and composition manufacturing tolerances are computed by taking the difference between the reference case and the case wherein the parameter associated with a manufacturing tolerance is at the tolerance limit that results in the increase in k_{eff}. The associated uncertainty is defined as the positive square root of the sum-of-the-squares of the Monte Carlo reference and "tolerance" case standard deviations times 1.65, the value of the 0.95 fractile of a standard normal distribution. Thus, for each component associated with a manufacturing tolerance, account is taken of the bias (applied only in the positive direction) and the computational uncertainty.

This procedure of adding biases and uncertainties to the reference k_{eff}, an already extreme value, (i.e., at optimum moderation) results in the licensee's "maximum (worst-case)" k_{eff}. The value obtained for k_{eff} in this manner for the NAPS fresh fuel storage area with fuel enriched to 4.6 wt % U^{235} is 0.93598. This is lower than the staff's acceptance criteria for k_{eff} of 0.95 for dry condition and 0.98 for optimum moderation.

In Refs. 4 and 5 and the submittal under evaluation, the expression "k_{eff} <u>on a 95/95 basis</u>" is used. The concept "on a 95/95 basis" originates in statistical sampling theory and expresses the limitation of an inference from a generally small sample. In the context of the analysis at hand, this is appropriate for the estimation of the methods bias, since the calculated k_{eff} is compared to a measured k_{eff} for a limited sample (59 cases) of critical experiments and an appropriate 95/95 tolerance coefficient for a sample size of 59 can be computed. However, the further implication that a k_{eff} value computed via Monte Carlo at the 95/95 tolerance limit of a mechanical or material composition parameter is also a 95/95 tolerance interval is not appropriate. Moreover, the 95/95 concept loses its meaning when biases and the root of the sum-of-the-squares of standard deviations are added to the nominal computed $k_{eff.}$

The licensee computes a so-called worst-case k_{eff} by taking into account uncertainties and biases in the computed system k_{eff} through the addition of a computational bias (determined by comparison to critical experiments) and a term for total uncertainties. The latter is the root mean square of the difference between the Monte Carlo computed k_{eff} for a reference case and ones where the geometry or composition have been perturbed.

To this is added 1.65 times the positive square root of the sum-of-the-squares of the Monte Carlo reference and “tolerance” case standard deviations. The justification for the use of 1.65, the one-sided 95th percentile point of a standard normal distribution, rather than a tolerance factor, is that the sample size for a Monte Carlo calculation is on the order of thousands, and can be considered infinite for practical applications. Thus, the application of 1.65 implies 100% confidence. The probabilistic interpretation of the final k_{eff} is not clear, other than that it is a very conservative “worst”-case value. Because the staff considers the k_{eff} computed to be a worst-case value, we find the analysis acceptable for demonstrating conformance to TS (Ref. 10) with regard to fresh or spent fuel storage rack k_{eff}.

2.2.2 Spent Fuel Storage

In addition to the considerations of geometry and enrichment that affect the fresh fuel criticality analysis, in the case of spent fuel storage, the degree of fissile depletion due to fuel burnup is taken into account since it reduces the reactivity worth of the spent fuel relative to its worth as fresh fuel, and allows more compact and, thereby, more economical fuel storage.

With regard to spent fuel, there are three basic conditions that are to be met so as to assure conformance with GDC-62: (Refs. 3, 4, and 5)

a) A k_{eff} < 1.0 assuming no soluble boron in the SFP.

b) A soluble boron concentration sufficient to ensure k_{eff} < 0.95.

c) An additional amount of soluble boron sufficient to offset the maximum reactivity effects of postulated accidents and to account for the uncertainty in the computed equivalence relation between the reactivity of fresh fuel and spent fuel.

The analytic methods used by the licensee to demonstrate these conditions are those prescribed in Ref. 4. The criticality calculations are performed with the KENO-V.a Monte Carlo code in x-y geometry and 238 energy groups. The axial burnup effect is taken into account through three-dimensional KENO calculations in 44 groups, with a pair of three-dimensional KENO calculations in 238 energy groups to verify the adequacy of the 44 group approximation in this context. The burnup credit curves and the sensitivity of k_{eff} to variations in material characteristics and mechanical/manufacturing dimensions for quantifying the associated uncertainties are computed with the PHOENIX code (Ref. 4), which is a depletable, two-dimensional, multi-group, discrete-ordinates transport theory code.

For storing spent fuel, two types of matrix arrangements of fuel assemblies are used. One consists of a 3 x 3 matrix of assemblies, and the other of a 5 x 5 matrix of assemblies. The licensee proposes the following loading restrictions as part of the administrative controls for storing spent fuel in racks. In the 3 x 3 cluster spent fuel assemblies with a reactivity equivalent to 2.0 wt % U^{235}, new fuel may be stored.

In the 5 x 5 cluster, the central cell is to contain no fuel. In the four cells surrounding the central cell, spent fuel of reactivity equivalent to 4.6 wt % U^{235} fresh fuel is allowed; and in the remaining cells, spent fuel of reactivity equivalent to 1.56 wt % U^{235} is allowed.

For the case of no soluble boron in the SFP, a k_{eff} value, including bias and uncertainties, of 0.99454 is computed for a 3 x 3 matrix configuration of fuel assemblies with reactivity equivalent to 2.0 wt % U^{235} new fuel. For the 5 x 5 matrix configuration with three different reactivity equivalent fuel assembly classes distributed as described above, the computed k_{eff}, with bias and uncertainties, is 0.99923. Both configurations meet the $k_{eff} < 1.0$ specification (Ref. 5).

For normal conditions, a computed concentration of soluble boron of 230 ppm is sufficient to maintain the SFP at a k_{eff} below 0.95 with bias and uncertainties taken into account. The computed k_{eff} under these conditions is 0.94908.

The fuel misplacement event is determined to be the limiting accident. The most limiting fuel misplacement event is a fresh fuel assembly of maximum-allowable enrichment placed in the center cell of the 5 x 5 cell matrix. For this event, an additional boron concentration of 550 ppm is required to maintain the spent fuel storage rack k_{eff} below 0.95, including bias and uncertainties.

Since the above KENO calculations of the spent fuel storage rack k_{eff} are based on fresh fuel, the administrative rule, which assigns spent fuel assemblies to a specific rack location based on the computed reactivity equivalence between a discharge burnup and a decrement in initial fuel enrichment, introduces additional uncertainty with regard to burnup measurement and modeling uncertainties of the PHOENIX code. These have been estimated as 0.0109 Δk and 0.0146 Δk respectively. To offset this additional uncertainty requires an additional soluble boron concentration of 120 ppm.

Thus, the total soluble boron credit required to ensure a $k_{eff} < 0.95$ is 900 ppm, the components of which are 230 ppm for normal conditions, 550 ppm for accident conditions, and 120 ppm to account for burnup equivalence uncertainty. As a safety margin, the licensee added 300 ppm of boron, making 1200 ppm the minimum boron concentration needed to satisfy the reactivity requirement in the SFP. Under the proposed TS, the boron concentration in the SFP will be maintained at a minimum of 2500 ppm; this is significantly above the 900 ppm needed to maintain $k_{eff} < 0.95$ as discussed above. The staff finds the proposed TS changes acceptable.

2.2.3 Storage of Other than North Anna Fuel Assemblies

The storage of Surry spent fuel assemblies in the NAPS SFP is permissible under the current NAPS operating license for spent fuel of up to 4.1 wt % U^{235} initial enrichment. Criticality analysis shows that Surry fuel is less reactive than NAPS fuel assemblies at the proposed enrichment of 4.6 wt % U^{235}. Thus, storage of Surry fuel in the high reactivity storage locations in the 5 x 5 matrix configuration is acceptable.

Additional fuel components such as fuel rod storage baskets, containers of loose fuel pellets and fuel pin fragments, and in-core detectors produce a k_{eff} less than the k_{eff} of a spent fuel assembly with an enrichment of 1.56 wt % equivalent U^{235} new fuel, and may, therefore, be stored in 5 x 5 matrix or 3 x 3 non-matrix locations.

2.2.4 Radiological Consequences of Accidents

Although the increase in the average discharge burnup of the NAPS fuel (with the increase in the maximum enrichment to 4.6 wt % U^{235}) will increase the radiological impact of an accident, numerous studies have shown that such an increase, as is proposed in this TS change, will meet the guidelines of 10 CFR Part 100 (Ref. 9). Therefore, the staff finds the proposed TS changes acceptable.

2.3 Boron Dilution Analysis and Decay Heat Removal Effects

In Westinghouse report WCAP-14416-A (Ref. 4), the methodology for crediting soluble boron in the SFP water when performing storage rack critically analysis for Westinghouse fuel storage pools is described (as approved in NRC evaluation of October 25, 1996 (Ref. 11)). Using this methodology, the licensee performed a boron dilution analysis to ensure that sufficient time is available to detect and mitigate a dilution event prior to exceeding the 0.95 k_{eff} design basis. In addition, the licensee did not take credit for the Boraflex neutron-absorbing material and assumed that the criticality was controlled only by the boron dissolved in the SFP. The licensee evaluated:

(1) SFP and related system features

  a. Dilution sources
  b. Dilution flow rates
  c. Boration sources
  d. Instrumentation
  e. Administrative procedures
  f. Piping
  g. Loss of offsite power impact

(2) Boron dilution initiating events (including operator error)

(3) Boron dilution times and volumes.

The above analysis was provided as part of the licensee's September 27, 2000, March 2, 2001, and May 21, 2001, submittals. In addition, as part of the criticality analysis for the SFP, the licensee eliminated credit for the presence of Boraflex in the SFP racks and relied entirely on soluble boron in the SFP.

2.3.1 Boron Dilution Analysis

The current NAPS TS do not include a requirement for SFP boron concentration. The proposed TS change, which credits soluble boron in the SFP water when performing storage rack criticality analysis, includes a new TS requirement for "Fuel Storage Boron Concentration" as discussed in an NRC evaluation of October 25, 1996 (Ref. 11). The North Anna fuel storage boron concentration is proposed to be ≥2500 ppm. This concentration of dissolved boron would be the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

In order to ensure that the design basis k_{eff} of 0.95 was not exceeded due to potential dilution events, the licensee determined that a boron concentration of 900 ppm would provide a k_{eff} of 0.95. This concentration includes 230 ppm for normal conditions, 550 ppm for accident conditions, and 120 ppm to account for burnup equivalence uncertainty. The licensee then evaluated various events that could potentially dilute the SFP.

NAPS, Units 1 and 2, share the same SFP, which has a water inventory of 409,695 gallons. The dilution analysis is based on an initial SFP water boron concentration of 2300 ppm, which is more conservative than the proposed TS concentration limit of 2500 ppm. To dilute the SFP boron concentration to 1200 ppm, which is conservative compared to the 900 ppm of boron needed to maintain the SFP k_{eff} at 0.95 or below, would require the addition of a volume of 266,543 gallons of unborated water to the SFP. The various events that were considered included dilution from: (1) the chemical and volume control system (CVCS); (2) open demineralizer valves; (3) the component cooling water system; (4) primary-grade (PG) water; (5) the fire protection system; and (6) plant heating system lines. Other events, such as seismic events, pipe break, and loss of offsite power were also considered.

Based on the events evaluated, there were only three water storage sources that could provide the 266,542 gallons of water needed to dilute the SFP. The first source is the PG reactor makeup tanks, which have a combined volume of 360,000 gallons with a makeup source from the PG water system. The PG makeup tanks are connected to the SFP through manually operated valves on the CVCS, demineralizer, and the PG water system hose connections. However, the largest dilution rate would be 200 gpm, which would take over 22 hours to dilute the SFP.

The second source is the condensate storage tank (600,000 gallons), which is sufficient to dilute the SFP. Therefore, the most rapid dilution would occur through an SFP cooling heat exchanger tube leak that interfaces with the condensate storage tank. This event would require over 44 hours at 100 gpm to dilute the pool.

Lake Anna, the third and final source, is considered an infinite source of makeup to the SFP that is capable of providing the 266,542 gallons needed to dilute the SFP. Lake Anna interfaces with the fire protection system through the manually operated emergency SFP fill valve. The dilution rate through this valve would be 400 gpm, which would take 11 hours to dilute the SFP. The licensee identified that inadvertently leaving the emergency fill valve open on the fire protection line would provide the largest flow rate of the dilution sources. Additionally, the opening of this valve to the fire protection line provides an essentially infinite source capable of diluting the SFP without replenishment. However, pool high level alarms or operator rounds every 6 hours would allow intervention and mitigation of the dilution event prior to exceeding the 0.95 k_{eff} design basis.

The licensee also evaluated smaller lines and determined that it would take at least 127 hours to dilute the pool. Since a dilution from one of the smaller lines would take longer than

6 hours, it would be identified and terminated by plant personnel during rounds, if it had not been terminated earlier upon SFP alarms or indications in the control room from interfacing systems.

Since all dilution events evaluated take longer than 6 hours to dilute the SFP, these events would be detected by plant personnel during required rounds every 6 hours. Additionally, the licensee has administrative procedures in place to detect, mitigate and suppress potential SFP dilution events. In addition to the administrative requirements, the licensee samples the SFP every 7 days to detect low-flow long-term dilution events. This frequency is consistent with the standard TS for Westinghouse plants and is considered appropriate for this plant.

The licensee concluded that an unplanned or inadvertent event that would dilute the SFP is not credible for North Anna. The staff finds that the combination of the large volume of water required for a dilution event, flow rates and dilution times, licensee administrative requirements, and TS-controlled SFP concentration and 7-day sampling requirement would adequately detect a dilution event prior to k_{eff} reaching 0.95. Therefore, the analysis, administrative controls, and proposed TS requirement is acceptable for the SFP dilution aspects of the request, as well as elimination of credit for Boraflex and instead relying on the dissolved boron for preventing criticality in the SFP.

2.3.2 Increased Fuel Enrichment Effects on Decay Heat Removal

The staff evaluated the effects of the proposed increase in fuel enrichment from 4.3 to 4.6 wt % on the decay heat removal capacity of the SFP cooling system. The SFP cooling system is designed to maintain water clarity and remove decay heat from the SFP. The licensee stated that 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. In addition, the licensee, as part of the reload safety evaluation, checks the impact of the decay heat load on the SFP cooling system.

In their May 21, 2001 submittal, the licensee confirmed that the decay heat load analysis performed as part of each reload is bounded by the Final Safety Analysis Report (FSAR) Section 9.1. If the decay heat load is not bounded by this section, then a safety evaluation will be performed and Section 9.1 of the UFSAR will be updated as necessary. These bounding analyses ensure that the SFP cooling system is capable of handling any increases in decay heat resulting from changes in the fuel management scheme, operational power level/cycle, and burnup. On the basis of its evaluation, the staff concludes that the increase in fuel enrichment will have an insignificant or no impact on the SFP cooling system to meet its intended design function and is, therefore, acceptable.

2.4 Staff Assessment

In view of the above considerations, the staff finds the following proposed changes to the TS to be acceptable.

a. Increase the limit on the feed fuel enrichment from the current maximum of 4.3 wt % U^{235} to a maximum of 4.6 wt % U^{235}.

b. Establish TS LCO with regard to SFP boron concentration and administrative restrictions on fuel storage location based on burnup and enrichment to accommodate the increase in the maximum fuel enrichment.

c. Eliminate the value for the allowance for uncertainties in the calculation for k_{eff} in the SFP criticality calculation, as computed by the reviewed and approved methodology.

d. Eliminate the credit for Boraflex and rely upon the dissolved boron for preventing criticality in the SFP.

There is reasonable assurance that the new TS will result in facility operation within the acceptance criteria of the UFSAR, and the health and safety of the public will not be endangered. In addition, for the reasons set forth above, the staff concludes that the licensee provided valid justification that all credible boron dilution events will not lower the boron concentration below the value required for preventing criticality in the SFP. These proposed TS changes are, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on April 25, 2001 (66 FR 20840) for this amendment. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. NRC, "Standard Review Plan," NUREG-0800, Draft Revision 3, Section 4.3, "Nuclear Design," April 1996.

2. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985.

3. Title 10 of the *Code of Federal Regulations*, Part 50, Appendix A.

4. WCAP-14416-NP-A, Rev. 1, “Westinghouse Spent Fuel Rack Criticality Analysis Methodology,” W. D. Newmyer, November 1996.

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

6. NUREG/CR-0200, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” December 1984.

7. Leslie N. Hartz, Virginia Electric and Power Company, to NRC, “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; Request for Additional Information,” December 18, 2000.

8. “Experience With the Scale Criticality Safety Cross-Section Libraries,” NUREG/CR-6686, October 2000.

9. “Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors,” NUREG/CR-5009, February 1988.

10. Letter from William R. Matthews, Virginia Electric and Power Company, to NRC, “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 Corrected Tables,” Serial No. 01-051, Docket Nos. 50-338 and 50-339, February 2, 2001.

11. NRC letter to Mr. T. Greene, Westinghouse Owners Group, dated October 25, 1996, Enclosure: NRC Safety Evaluation of WCAP-14416-P.

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