UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

50-390

July 28, 1997

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: ISSUANCE OF AMENDMENT - SPENT FUEL POOL STORAGE CAPACITY INCREASE (TAC NO. M96930)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 6 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant (WBN), Unit 1. This amendment is in response to your application dated October 23, 1996, as supplemented December 11, 1996, January 31, February 10 and 24, March 11, April 4 and 11, May 28, June 26 and July 15, 1997. The amendment changes the WBN Technical Specifications (TS) to increase the spent fuel storage capacity from 484 fuel assemblies to 1610 fuel assemblies and to increase the initial enrichment of the fuel to be stored in the spent fuel storage racks from 3.5 weight percent (wt%) to 5.0 wt%. This modification also changes the center-to-center spacing of stored fuel assemblies and reflects the use of burnup credit rack modules to be installed peripherally along the pool walls. A limiting condition for operation will be added to require that the combination of initial enrichment and burnup of each spent fuel assembly to be stored is in the acceptable region.

The amendment, as proposed by the licensee, would also involve the installation of spent fuel racks in the spent fuel cask pit for 225 storage spaces thus increasing the total WBN spent fuel storage capacity to 1835 spent fuel assemblies. The licensee proposed to provide an impact shield that would be placed over the fuel in the cask pit when heavy loads are moved near or across the cask pit area. The staff is continuing its review of this aspect of the licensee's proposal. Accordingly, this amendment authorizes the reracking and usage of the main spent fuel pool, as proposed for a total of 1610 spent fuel spaces. However, it does not authorize the installation of storage racks or storage of spent fuel in the spent fuel cask pit. The staff's review of the spent fuel cask aspects of the licensee's application will be addressed by further correspondence.

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A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

original signed by R.Martin

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures: 1. Amendment No. 6 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

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A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-390

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2. Safety Evaluation

cc w/enclosures: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. NPF-90

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 23, 1996, as supplemented December 11, 1996, January 31, February 10 and 24, March 11, April 4 and 11, May 28, June 26, and July 15, 1997, 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 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-90 is hereby amended to read as follows:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 6 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented no later than completion of the reracking modification or prior to the movement of fuel assemblies into the spent fuel pool for the Cycle 1 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 28, 1997

ATTACHMENT TO AMENDMENT NO. 6

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

| Remove Pages | Insert Pages |
|---|---|
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| v | v |
| iii of Bases | iii of Bases |
| - | 3.7-31 |
| - | 3.7-32 |
| - | B 3.7-75 |
| - | B 3.7-76 |
| - | B 3.7-77 |
| B 3.9-33 | B 3.9-33 |
| 4.0-2 | 4.0-2 |
| 4.0-3 | 4.0-3 |
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| 4.0-5 | 4.0-5 |
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## 3.7 PLANT SYSTEMS

### 3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 1 or Region 2 shall be within the Acceptable Burnup Domain of Figure 3.7.15-1 or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.15.1 | Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 or Specification 4.3.1.1. | Prior to storing the fuel assembly. |

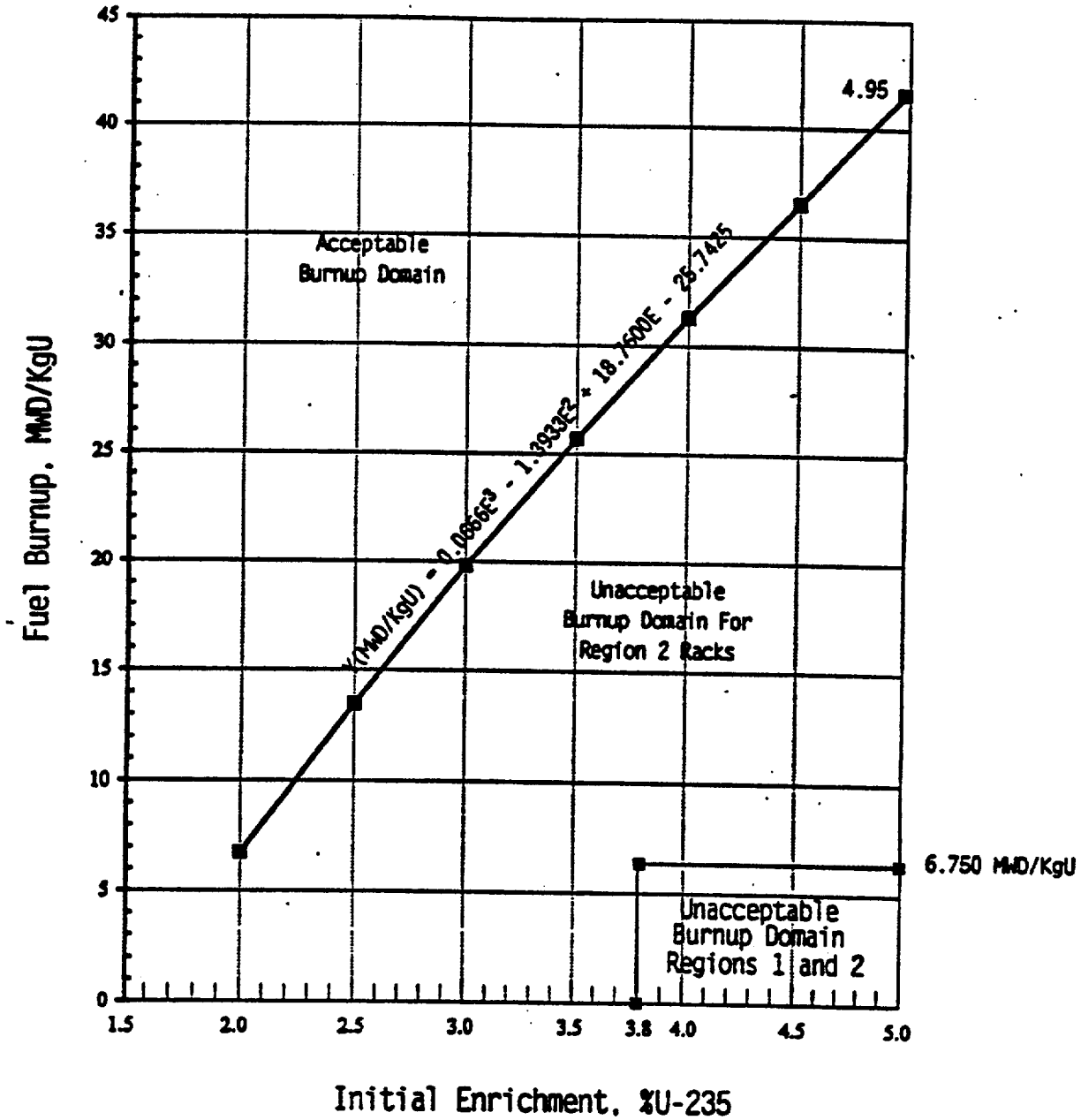


Figure 3.7.15-1 Acceptable Burnup Domain - Watts Bar Spent Fuel Storage Racks

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

In the spent fuel storage design (References 1 and 2), the spent fuel pool area is divided into two separate and distinct regions for criticality considerations. Region 1, flux trap modules with 1386 storage positions, is designed to accommodate fuel with enrichment as high as 3.8 weight percent U-235 without restrictions. Storage of fuel assemblies with enrichment between 3.8 and 5.0 weight percent in Region 1 requires either fuel burnup of ≥6.750 MWD/KgU in accordance with Figure 3.7.15-1, or placement in storage locations which have face adjacent storage cells containing either water or fuel assemblies with accumulated burnup of at least 20.0 MWD/KgU in accordance with paragraph 4.3.1.1.

Region 2 burnup credit rack modules, with 224 total storage positions, is designed to accommodate fuel with 4.95 ± 0.05 weight percent initial enrichment burned to at least 41 MWD/KgU or, in accordance with Figure 3.7.15-1, fuel of lower enrichment which yields an equivalent reactivity.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting $k_{eff}$ of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975, and the April 1978 NRC letter (Reference 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, an

BASES

BACKGROUND (continued)

abnormal scenario could be associated with the improper movement of a relatively high enrichment, low exposure fuel assembly from Region 1 to Region 2, or the misloading of a fuel assembly in either region. This could potentially increase the criticality of the storage regions. To mitigate these postulated criticality-related events, boron is dissolved in the pool water. Safe operation of the spent fuel storage design with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly in the pool, it is necessary to perform SR 3.9.9.1.

APPLICABLE SAFETY ANALYSES

The hypothetical events can only take place during or as a result of the movement of an assembly. For these occurrences, the presence of soluble boron in the spent fuel storage pool, (controlled by LCO 3.9.9, "Spent Fuel Pool Boron Concentration,") prevents criticality in both storage rack regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential occurrences may be limited to a small fraction of the total operating time. During the remaining time period with no potential for such events, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool in accordance with Figure 3.7.15-1, in the accompanying LCO, ensures the $k_{eff}$ will always remain $\leq 0.95$, assuming the pool to be flooded with unborated water. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.

BASES (continued)

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If unable to move irradiated fuel assemblies while in Mode 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in Mode 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with Figure 3.7.15-1, or paragraph 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movements to bring the configuration into compliance with Figure 3.7.15-1 or Specification 4.3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.15-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. Watts Bar FSAR, Sections 4.3.2.7 and 9.1.2.

2. Spent Fuel Pool Modification for Increased Storage Capacity, (Chapter 4), Watts Bar Unit 1, submitted by TVA letter dated October 23, 1996.

3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

## B 3.9 REFUELING OPERATIONS

## B 3.9.9 Spent Fuel Pool Boron Concentration

BASES

| | |
|---|---|
| BACKGROUND | The spent fuel storage rack criticality analysis assumes 2000 ppm soluble boron in the fuel pool during a dropped/misplaced fuel assembly event. |
| APPLICABLE SAFETY ANALYSES | This requirement ensures the presence of at least 2000 ppm soluble boron in the spent fuel pool water as assumed in the spent fuel rack criticality analysis for dropped/misplaced fuel assembly event.<br><br>The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement. |
| LCO | The LCO requires that the boron concentration in the spent fuel pool be greater than or equal to 2000 ppm during fuel movement. |
| APPLICABILITY | This LCO is applicable when the spent fuel pool is flooded and fuel is being moved. Once fuel movement begins, the movement is considered in progress until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Figure 3.7.15-1 and Specification 4.3.1.1. |
| ACTIONS | A.1<br><br>If the spent fuel pool boron concentration does not meet the above requirements, fuel handling in the spent fuel pool must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis.<br><br>Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. |

## 4.0 DESIGN FEATURES

### 4.3 Fuel Storage

#### 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which, includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR;

c. Distances between fuel assemblies as follows:

1. A nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules (Region 1).

2. A nominal 8.972 inch center-to-center spacing in the ten burnup credit rack modules peripherally located adjacent to the south and west pool walls (Region 2); and

d. Spent fuel assemblies with a burnup in the "acceptable burnup domain" of Figure 3.7.15-1 may be allowed unrestricted storage in either type of fuel storage rack.

e. New or partially spent fuel assemblies with a burnup in the "unacceptable burnup domain" of Figure 3.7.15-1 will be stored in compliance with the following configuration:

(continued)

## 4.0 DESIGN FEATURES

### 4.3 Fuel Storage (continued)

1. In the flux trap rack modules (Region 1), fuel assemblies with enrichments less than or equal to 3.80 weight percent U-235 are allowed unrestricted storage. Fuel assemblies with enrichment greater than 3.80 weight percent U-235 and burnup less than 6.750 megawattday/kilogram uranium (MWD/KgU) shall be placed in storage cells that face adjacent cells in the flux trap modules containing either water or fuel assemblies with accumulated burnup of at least 20 MWD/KgU.

2. Storage in any burnup credit rack modules (Region 2) located in the pool is restricted to fuel of 4.95 ± 0.05 weight percent initial enrichment burned to at least 41 MWD/KgU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity in the fuel racks. The minimum required assembly average burnup is given by Y(MWD/KgU) where $Y = 0.0666E^3 - 1.3933E^2 + 18.7600E - 25.7425$, where E is the initial enrichment in the axial zone of highest enrichment. Figure 3.7.15-1 illustrates the burnup enrichment equation in graphical form.

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

(continued)

## 4.0 DESIGN FEATURES

### 4.3 Fuel Storage (continued)

a. Fuel assemblies having a maximum U-235 enrichment of 4.3 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Elevation 747 feet - 1 1/2 inches.

4.3.3 Capacity

The total spent fuel storage capacity is 1610 fuel assemblies.

4.3.3.1 The primary portion of the spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies in 24 flux trap rack modules.

4.3.3.2 No more than 224 fuel assemblies will be stored in ten smaller burnup credit rack modules peripherally located adjacent to the south and west walls of the pool.

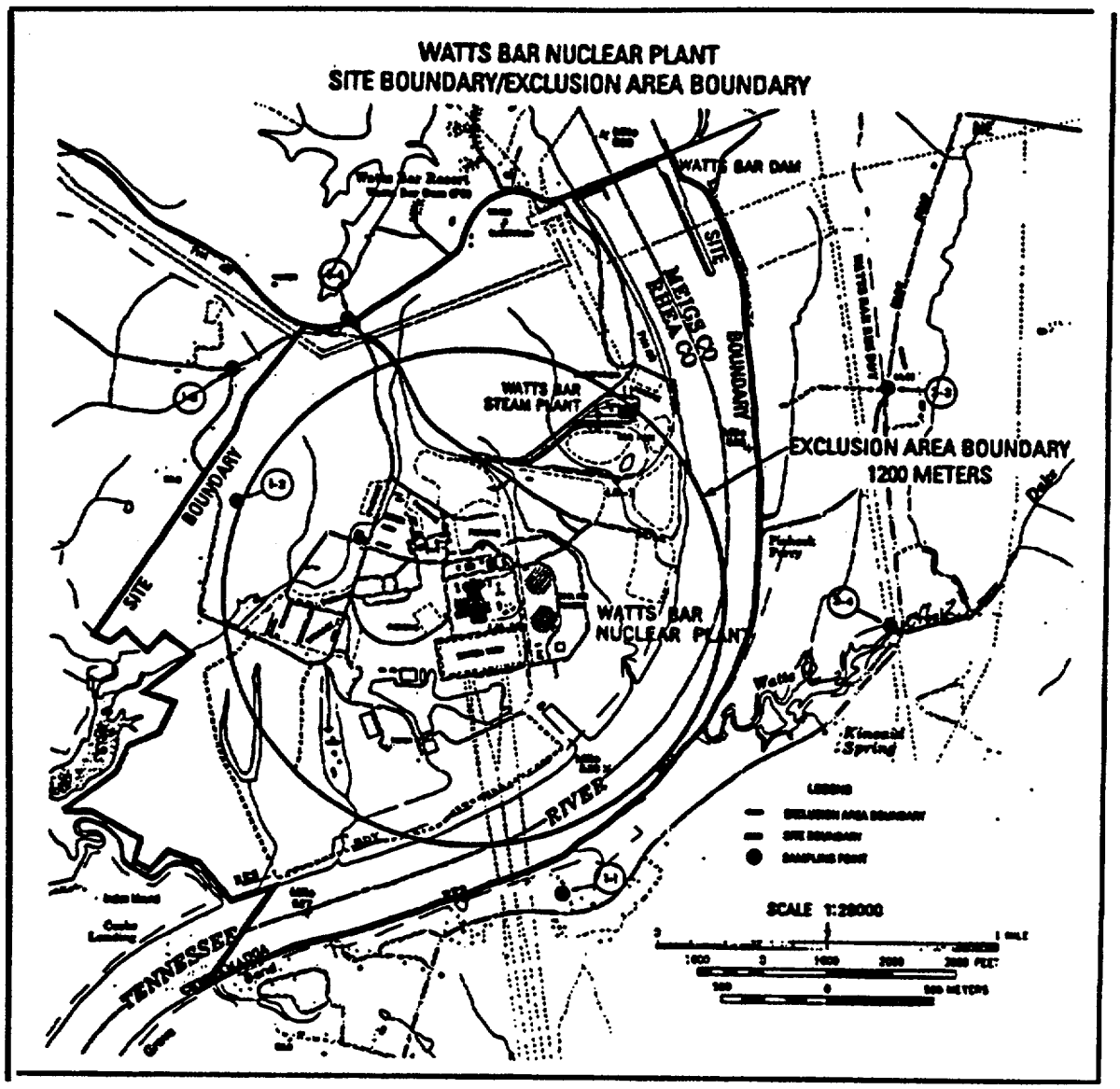


Figure 4.1-1 (page 1 of 1)
Site and Exclusion Area Boundaries

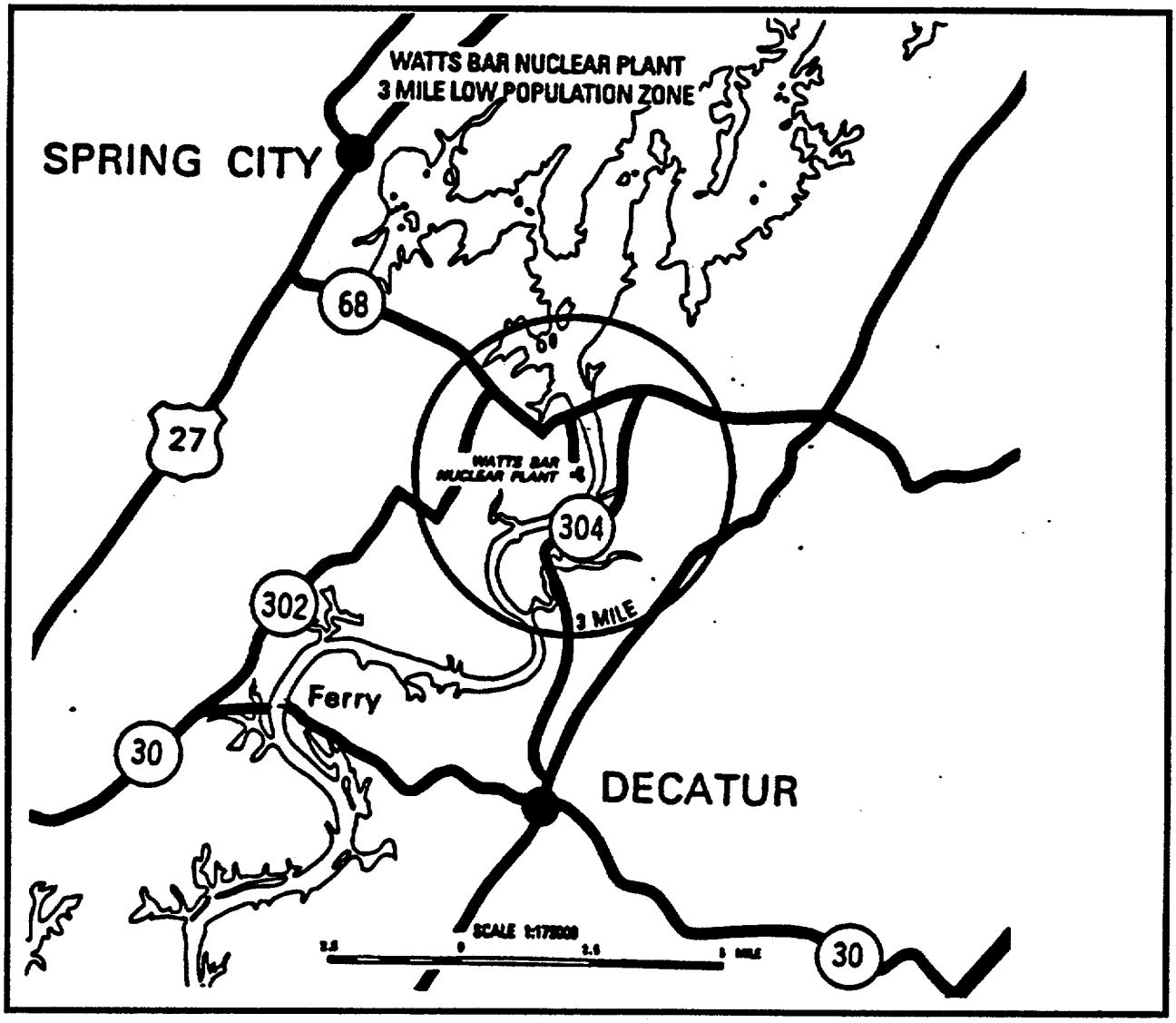


Figure 4.1-2 (page 1 of 1)
Low Population Zone

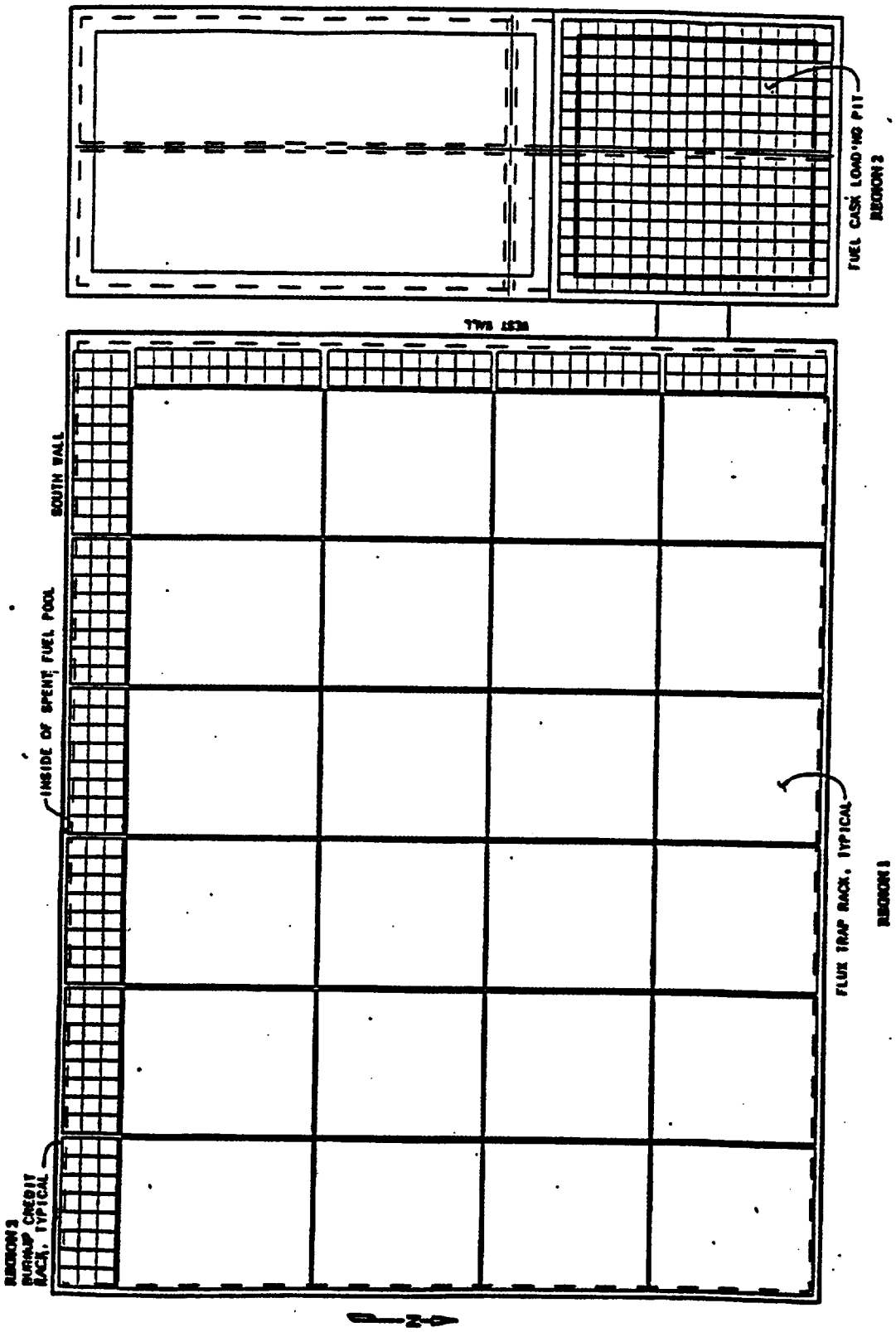


FIGURE 4.3-1
SPENT FUEL STORAGE RACKS

PLAN
SPENT FUEL POOL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

## 1.0 INTRODUCTION

By letter dated October 23, 1996, as supplemented December 11, 1996, January 31, February 10 and 24, March 11, April 4 and 11, May 28, June 26 and July 15, 1997, the Tennessee Valley Authority (TVA) submitted a request for changes to the Watts Bar Nuclear Plant (WBN), Unit 1, Technical Specifications (TS). The racks originally installed in WBN contained 1312 fuel storage locations of which only 484 were usable due to concerns about the Boraflex neutron absorber and welding fabrication, as reviewed in supplements to the licensing Safety Evaluation Report, NUREG-0847, for WBN. This amendment changes the WBN TS to increase the spent fuel storage capacity from 484 fuel assemblies to 1610 fuel assemblies and to increase the initial enrichment of the fuel to be stored in the spent fuel storage racks from 3.5 weight percent (wt%) to 5.0 wt%. This modification also changes the center-to-center spacing of stored fuel assemblies and reflects the use of burnup credit rack modules to be installed peripherally along the pool walls. A limiting condition for operation will be added to require that the combination of initial enrichment and burnup of each spent fuel assembly to be stored is in the acceptable region.

The amendment, as proposed by the licensee, would also involve the installation of spent fuel racks in the spent fuel cask pit for 225 storage spaces thus increasing the total WBN spent fuel storage capacity to 1835 spent fuel assemblies. The licensee proposed to provide an impact shield that would be placed over the fuel in the cask pit when heavy loads are moved near or across the cask pit area. The staff is continuing its review of this aspect of the licensee's proposal. Accordingly, this amendment authorizes the reracking and usage of the main spent fuel pool, as proposed for a total of 1610 spent fuel spaces. However, it does not authorize the installation of storage racks or storage of spent fuel in the spent fuel cask pit. The staff's review of the spent fuel cask pit aspects of the licensee's application will be addressed by further correspondence.

TVA proposed to increase the capacity of the spent fuel pool by replacing the existing fuel racks. Some of the replacement racks will come from the Sequoyah plant (SQN) and some will be newly designed and constructed by Holtec International for WBN. All of the replacement fuel racks will use Boral

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neutron absorber instead of Boraflex which was used in the originally installed WBN racks. TVA plans to install two types of spent fuel racks:

- Twenty-four modules containing 1386 storage cells, manufactured by Programmed and Remote System Corporation (PaR) and used previously for about 13 years in the SQN plant

- Ten smaller modules ("baby" racks), containing 224 storage cells manufactured by Holtec International

The spacing of the fuel assembly center-to-center distance will change from a nominal 10.72 inches to 10.375 inches in the 24 PaR flux trap rack modules positioned in the spent fuel pool in a free-standing four-by-six module array. The fuel assembly center-to-center distance will be 8.972 inches in the 10 smaller burnup credit rack modules containing 224 cells with Boral to be installed peripherally along the south and west walls of the pool. The reracking would increase the current storage capacity of 484 assemblies to a storage capacity of 1610 assemblies.

A limiting condition for operation is added to require that the combination of initial enrichment and burnup of each spent fuel assembly to be stored is in the acceptable region.

The April 4 and 11, May 28, June 26, and July 15, 1997 letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

## 2.0 EVALUATION

### 2.1 Criticality

As stated in TVA's letter of October 23, 1996 the current TS 4.3.1.2 for fresh (unirradiated) fuel storage in the new fuel storage area is limited to 4.3 wt% U-235 enriched fuel. TVA does not anticipate fresh fuel with enrichment greater than 4.3 wt% U-235 to be stored in the new fuel storage area before fuel is purchased for Cycle 3. TVA's response dated December 11, 1996, stated that TVA will request a license amendment to TS 4.3.1.2 before fuel greater than 4.3 wt% U-235 is received. The staff's evaluation of the criticality aspects of the proposed rerack follows.

The 24 replacement racks containing 1386 storage cells were designed and manufactured by Programmed and Remote System Corporation (PaR) in 1979 for TVA's Sequoyah Nuclear Plant and were licensed and used successfully in service at that facility for about 13 years until their replacement in 1995 with higher density racks. The smaller burnup credit racks were designed by Holtec International and are identical in design and construction to the Holtec rack cells presently installed and in use at the Sequoyah plant.

The analysis of the reactivity effects of fuel storage in the WBN spent fuel pool was performed with the three-dimensional Monte Carlo code, KENO-5. Since the KENO-5 code package does not have burnup capability, depletion analyses were made with the two-dimensional integral transport theory code, CASMO-3. CASMO-3 was also used for the determination of small reactivity increments due to manufacturing tolerances in the Region 2 analysis. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These

experiments simulate the WBN spent fuel racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and neutron absorber worth. A sufficient number of neutron histories were accumulated in each calculation to minimize the statistical uncertainty of the KENO-5 calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the WBN storage racks with a high degree of confidence.

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These assumptions included ignoring (1) radial neutron leakage from the racks, (2) the presence of control rods, (3) the presence of spent burnable absorber assemblies, and (4) the reactivity decrease due to higher water temperatures. The design basis fuel assembly was a Westinghouse VANTAGE 5H design. Calculations have shown that this is the most reactive fuel assembly design at WBN for the maximum enrichment considered.

Based on the above, the staff concludes that appropriately conservative assumptions were made.

General Design Criterion (GDC) 62 of Appendix A to 10 CFR Part 50 states that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. This requirement is met by conforming to the U.S. Nuclear Regulatory Commission (NRC) acceptance criterion for criticality which states that the effective neutron multiplication factor ($k_{eff}$) in the spent fuel pool storage racks, if fully flooded by unborated water, shall be no greater than 0.95, including uncertainties at a 95/95 probability/confidence level.

For the Region 1 analysis, biases due to the calculational method, the enrichment extrapolation from critical experiments, boron particle self-shielding, and, where applicable, a temperature correction to 4°C were included. Uncertainties due to the KENO-5 statistics and the KENO-5 method and enrichment, as well as burnup, where applicable, were statistically combined with mechanical tolerance uncertainties. An evaluation of the uncertainties due to dimensional tolerances (cell bow, cell pitch, inner stainless steel wall, outer stainless steel wall, inner tube inside width, outer tube inside width, and poison plate width) showed that the most significant adverse reactivity effect was due to the combined impact on the flux trap. Therefore, a worst-case analysis which minimized the flux trap width was used to calculate the reactivity impact of mechanical tolerances. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

The licensee's analysis of the Region 1 storage racks showed that fuel with 3.8 wt% U-235 enrichment results in a maximum $k_{eff}$ of 0.9456 including biases and 95/95 uncertainties. Fresh (unirradiated) fuel with 5.0 wt% (4.95 + 0.05) U-235 enrichment alternating with water channels in a checkerboard configuration results in a maximum $k_{eff}$ of 0.8673 including biases and 95/95 uncertainties.

The concept of burnup reactivity equivalencing was also used in order to store fuel with nominal enrichment greater than 3.8 wt% U-235, but no greater than 5.0 wt% U-235. This concept is based on the reactivity decrease associated with fuel depletion and has been previously found acceptable by the NRC for

use in pressurized water reactor fuel storage analysis. A series of reactivity calculations is performed to generate a set of enrichment versus burnup ordered pairs which yield an equivalent $k_{eff}$ for fuel stored in the WBN racks. The results of the burnup reactivity equivalencing shows that fuel with an initial U-235 enrichment of 5.0 wt% and irradiated to 6.75 MWD/KgU results in a maximum $k_{eff}$ of 0.9477 including biases and 95/95 uncertainties. In addition, fresh fuel with a 5.0 wt% U-235 enrichment checkerboarded with fuel assemblies irradiated to 20 MWD/KgU results in a maximum $k_{eff}$ of 0.9457 including biases and 95/95 uncertainties.

The results of these analyses, using the acceptable methods discussed above, meet the NRC criterion of $k_{eff}$ no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level and are, therefore, acceptable. The results show that Region 1 can accommodate fuel with enrichment as high as 3.8 wt% U-235 without restriction. Storage of fuel assemblies with enrichment between 3.8 and 5.0 wt% U-235 requires either fuel burnup of at least 6.750 MWD/KgU in accordance with TS Figure 3.7.15-1, or placement in storage locations which have face adjacent storage cells containing either water or fuel assemblies with accumulated burnup of at least 20 MWD/KgU.

Although not included in the burnup dependent criticality analyses, subsequent decay of Pu-241 with long-term storage results in a significant decrease in reactivity. This will provide an increasing subcriticality margin and further compensate for any uncertainty in the depletion calculations.

The Region 2 criticality analysis was performed by Holtec. For the Region 2 analysis, biases due to the calculational method and uncertainty in the depletion calculations were included. Uncertainties due to the KENO-5 statistics and the KENO-5 method, as well as burnup, where applicable, were statistically combined with mechanical tolerance uncertainties. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

The Holtec analysis of the Region 2 storage racks showed that fuel with an initial enrichment of 5.0 wt% U-235 that has accumulated burnup of at least 41 MWD/KgU results in a maximum $k_{eff}$ of 0.9444 including biases and 95/95 uncertainties. Fuel with enrichment lower than 5.0 wt% U-235 that has accumulated burnup in the acceptable burnup domain of TS Figure 3.7.15-1 may also be stored in the Region 2. This acceptable domain is represented by the following equation, where E is the enrichment in wt% U-235:

$$\text{Burnup (MWD/KgU)} = 0.0666E^3 - 1.3933E^2 + 18.7600E - 25.7425$$

The results of these analyses, using the acceptable methods discussed above, meet the NRC criterion of $k_{eff}$ no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level, and are therefore, acceptable.

A water gap of 1.5 inches between Region 1 and Region 2 racks, with Boral panels on both sides, precludes any adverse interaction between the two regions.

Most abnormal storage conditions will not result in an increase in the $k_{eff}$ of the racks. However, it is possible to postulate events, such as the

inadvertent misloading of an assembly with a burnup and enrichment combination outside of the acceptable areas in TS Figure 3.7.15-1, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of at least 2000 ppm of soluble boron in the pool which is required by TS 3.9.9 and 3.9.10 since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The reduction in $k_{eff}$ caused by the boron more than offsets the reactivity addition caused by credible accidents.

The following TS changes have been proposed as a result of the requested spent fuel pool reracking. Based on the above evaluation, the staff finds these changes acceptable as well as the associated Bases changes.

(1) TS 4.3.1.1 has been revised to allow an increase in the maximum initial enrichment of fuel to be stored in the spent fuel storage racks from 3.50 wt% to 5.0 wt% U-235. The revision would also allow a change in the spacing of stored fuel assemblies from a nominal 10.72-inch center-to-center distance to nominal center-to-center distances of 10.375 inches in the 24 PaR flux trap rack modules to be installed in the pool. A nominal 8.972-inch center-to-center distance would also be allowed for the smaller Holtec burnup credit racks to be installed peripherally along the south and west pool walls. The revision also specifies fuel assembly loading arrangements in both types of rack modules which relate locations to the initial enrichment and burnup provisions given in Figure 3.7.15-1.

(2) TS 4.3.3 has been modified to reflect the increased fuel pool storage capacity from 484 to 1610 fuel assemblies. Figure 4.3-1 is also added to illustrate the layout of the rack modules in the pool.

(3) TS 3.7.15 has been modified to require that the combination of initial enrichment and burnup of each stored assembly be in an acceptable burnup domain. Figure 3.7.15-1 illustrates the relationship between fuel assembly burnup and initial enrichment which will be applicable.

2.1.1 Criticality - Conclusion

Based on the review described above, the staff finds the criticality aspects of the proposed increase in the storage capacity of the WBN spent fuel pool storage racks are acceptable and meet the requirements of GDC-62 for the prevention of criticality in fuel storage and handling.

Although the WBN TS have been modified to specify the above-mentioned fuel as acceptable for storage in the spent fuel racks, evaluations of reload core designs (using any enrichment) will, of course, be performed on a cycle-by-cycle basis as part of the reload safety evaluation process. Each reload design is evaluated to confirm that the cycle core design adheres to the limits that exist in the accident analyses and TS to ensure that reactor operation is acceptable.

## 2.2 Plant Systems

### 2.2.1 Introduction

The following evaluation covers the applicable areas of the TVA submittals dated February 10, April 4, and June 25, 1997 in the plant systems area of review.

### 2.2.2 Spent Fuel Pool (SFP) Cooling and Cleaning System

The SFP cooling and cleaning system (SFPCCS), which contains two subsystems (SFP cooling system and SFP cleaning system), is designed to remove the decay heat generated by stored spent fuel assemblies (SFAs), and to clarify and purify the water in the SFP, transfer canal, and refueling storage tanks. The cleaning system portion of the SFPCCS can be isolated by manual valves from the spent fuel pool cooling system (SFPCS) portion of the SFPCCS.

The SFPCS consists primarily of two Quality Group B and Seismic Category 1 cooling trains, each equipped with one heat exchanger and one pump. A third identical pump is provided and installed to serve as backup to the pump in either cooling train of the SFPCS. Heat is removed from the SFP heat exchangers by the component cooling water system. Electrical power to each of the SFP cooling pumps (including the backup pump) is supplied from emergency power buses.

The SFPCS was initially designed with two trains to maintain the SFP water temperature at or below 120°F during routine refueling outages with an SFP heat load of $23.77 \times 10^6$ Btu/hr. An SFP water temperature of 150°F could also be maintained with one train in operation and an SFP heat load of $26.27 \times 10^6$ Btu/hr resulting from an unplanned full core discharge subsequent to a routine refueling outage.

As a result of an increase in the SFAs planned to be stored in the SFP and plans to conduct a full core discharge as a part of each routine refueling outage, the decay heat load for any specific fuel discharge scenario will increase. The licensee performed an analysis for the following discharge scenarios to evaluate the effects of increased SFP storage capacity on the SFPCS and to establish a bounding heat load capacity for the SFP.

The following is the revised decay heat generated in the SFP and the corresponding calculated peak SFP temperatures for various scenarios with one or two SFP cooling trains in operation:

| | Peak SFP Temp. | Coincident SFP Loads ($10^6$ Btu/Hr) |
|---|---|---|
| Case 1 (2 Cooling Trains) | 124.69°F | 28.10 |
| (1 Cooling Train) | 151.17°F | 27.90 |
| Case 2 (2 Cooling Trains) | 129.30°F | 32.60 |
| (1 Cooling Train) | 159.64°F | 32.41 |

For the SFP heat load during the routine[1] refueling outage (Case 1), the licensee assumed that 1680 SFAs were stored in the SFP at the start of the final discharge. For the SFP heat load resulting from an unplanned full core off-load outage (Case 2), the licensee assumed that the SFP is at the "end-of-pool" lifetime. A total of 1873[2] SFAs, which exceeded the analysis capacity (1835 SFAs) of the SFP, were assumed to be stored in the SFP.

As indicated above, the increase in spent fuel storage will result in an increase in the SFP heat load (from $23.77 \times 10^6$ Btu/hr to approximately $28 \times 10^6$ Btu/hr) during routine refueling outages. The peak SFP water temperature resulting from this heat load increase is 124.69°F with two SFP cooling trains in operation. With one SFP cooling train in operation, the peak SFP water temperature is 151.17°F.

The decay heat load during an unplanned full core off-load outage will increase from $26.27 \times 10^6$ Btu/hr to slightly over $32 \times 10^6$ Btu/hr. The corresponding calculated peak SFP bulk temperature will be 159.24°F with one cooling train operating and 129.3°F with both cooling trains operating. These temperatures are well below the guidance of the Standard Review Plan (SRP) for the SFP water temperature limit (pool boiling) during an unplanned full core off-load outage.

The above revised decay heat loads during routine refueling and unplanned full core off-load outages were calculated by the licensee's contractor, Holtec International, using the ORIGEN2 computer code originally developed by ORNL for decay heat calculation. The decay heat load in the SFP during unplanned full core off-load outages is $32.60 \times 10^6$ Btu/hr. In submittal dated June 25, 1997, the licensee established $32.60 \times 10^6$ Btu/hr as the bounding heat load in the SFP for planned and unplanned outages. The licensee stated that their approach for decay heat removal during routine refueling outages is to evaluate the decay heat loads and the decay heat removal systems available for each refueling outage. Prior to each refueling or unplanned core off-load outage, WBN plant operating procedures require the TVA Nuclear Fuel Group to calculate the decay heat loads for the WBN core as a function of time after shutdown for the specific end-of-cycle core to be off-loaded and for any existing spent fuel assemblies in the pool. The time at which core off-load starts will be determined such that the design spent fuel pool decay heat load of $32.60 \times 10^6$ Btu/hr is not exceeded and in the worst case design basis single active failure scenario, the SFP temperature will not exceed 159°F. In any event, the in-vessel decay time must be equal to or greater than 100 hours as currently required by TVA's Technical Requirement (TR) 3.9.1.

For each cycle-specific core off-load, TVA uses the DHEAT computer code to determine the decay heat generation in the SFP. TVA stated that DHEAT computer code was developed by TVA based on 1979 American Nuclear Society

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[1] Full core discharge is planned to be a part of each routine refueling operation at the WBN plant. One full core off-load to the SFP starts 12 days after shutdown at a rate of four fuel assemblies per hour. 113 freshly discharged SFAs plus 80 new fuel assemblies are returned into the reactor approximately 30 days after off-load.

[2] 1680 SFAs (including 80 SFAs from a planned refueling outage having been previously discharged in the pool for 36 days) are stored in the SFP, followed by a full core (193 SFAs) discharge.

Standard 5.1, "Decay Heat Power in Light Water Reactors," and Regulatory Guide (RG) 3.54, "Spent Fuel Heat Generation in An Independent Spent Fuel Storage Installation," 1984. The code is controlled under the TVA Nuclear Software Quality Assurance Plan. Calculations are documented and controlled by procedures. The licensee further stated that the results from using DHEAT code compare very well with those results using NRC Branch Technical Position ASB 9-2, "Residual Decay Energy For Light-Water Reactors for Long-Term Cooling," Revision 2, for the same thermal power and cumulative reactor operating time. The staff has performed confirmatory decay heat load calculations and results of its confirmatory calculations agree with the licensee's revised decay heat calculation results.

In a submittal dated June 25, 1997, the licensee committed to the following:

* For planned and unplanned refueling outages, TVA will limit the heat load in SFP to $32.60 \times 10^6$ Btu/hr.

* TVA will ensure that sufficient SFP cooling equipment is operating and the rate of fuel transfer is controlled such that the SFP temperature does not exceed 150°F during anticipated refueling activities.

* In the worst case design basis single active failure scenario, the SFP temperature will not exceed 159°F.

* Operating procedures will provide the controls necessary to ensure the above limitations are met.

* A decay heat calculation is routinely performed at the end of each operating cycle to show decay heat vs. time curves for the core and SFP.

In addition, the SFP has a water temperature monitor which alarms in the control room when the SFP water temperature reaches 127°F. Annunciator response instructions list the probable causes and corrective actions to be taken when the high temperature alarm is received. This provides additional assurance that the above limitations are met.

Also, the licensee committed to update the plant's Final Safety Analysis Report (FSAR) to incorporate the five items listed above regarding the SFP heat load and temperature, and how fuel off-loads to the SFP are controlled.

Based on the staff's review and TVA's commitments described above, the staff finds that the design and operation of the SFPCS meets the intent of the SRP for SFPs and that the increase in peak calculated SFP water temperature resulting from the increase in the SFP decay heat load during a routine refueling or an unplanned full core off-load outage is acceptable.

### 2.2.3 Effects of SFP Boiling

In the event that there is a complete loss of cooling capability of using SFPCS heat exchangers to remove heat from the SFP, the SFP water temperature will begin to rise and eventually will reach the boiling temperature. The licensee performed analysis which demonstrates that the calculated minimum time from the loss-of-pool cooling until the pool boils is 5.86 hours. Once boiling starts, the maximum boil-off rate would be 68.59 gallons per minute (gpm). With this boil-off rate and the addition of 100°F water at 55 gpm at any time within 10 hours following the loss of cooling, the licensee

calculated that the minimum water level above the top of the fuel that would be reached would be approximately 21 feet. Based on this, the licensee determined that sufficient time is available to establish makeup water from a qualified source. As described in the WBN FSAR, makeup water to the pool can be provided by the demineralized water system, the refueling water storage tank (RWST), or the fire protection system to maintain the pool water level. All piping, valves, and pumps from the RWST to the SFP cooling loop are seismically qualified. The fire protection system is a Seismic Category I system.

Based on its review, the staff finds that cooling the SFP by allowing the SFP to boil and by adding makeup water in the event of a complete loss of cooling capability (using SFPCS heat exchangers to remove heat from the SFP) conforms with the guidance described in the SRP and, therefore, is acceptable.

2.2.4 Conclusion

Based on the staff's review of TVA's rationale, commitments described above, and the commitment to update the plant's FSAR to reflect the above information regarding the SFP and how fuel off-loads to the SFP are controlled, the staff concludes that the proposed TS changes to increase the SFP storage capacity from 484 to 1835 fuel assemblies at WBN with respect to the SFP cooling capacities are acceptable. Note, however, that this amendment authorizes an increase in the capacity only up to 1610 spaces as discussed in the introduction above.

Separately, an issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83 and its Supplement 1, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," dated October 7, 1993 and August 24, 1995, respectively, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the license under separate cover.

2.3 Structural Engineering

2.3.1 Introduction

This evaluation addresses the adequacy of the structural aspects of TVA's proposal to increase the spent fuel storage capacity at WBN as set forth in submittals dated October 23, 1996, February 24, May 28, June 26 and July 15, 1997. The primary purpose of this review is to assure the structural integrity of the rack modules, stored fuel assemblies, and the spent fuel pool structure under the postulated loads (Appendix D of SRP, Section 3.8.4) and fuel handling accidents.

2.3.2 Storage Racks

The storage cells will be contained in 35 fuel storage racks, which are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). Among the 35 storage rack modules, 24 were designed and manufactured by Programmed and Remote System Corporation (PaR) for TVA's Sequoyah Nuclear Plant (SQN) and were licensed and used in

service at SQN for about 13 years. Those rack modules have been transported to WBN from SQN and will be re-used at WBN. Eleven other rack modules are new, and are designed and manufactured by Holtec. TVA performed structural analyses for the Holtec racks along with reanalyses for the PaR racks for the requested license amendment.

TVA used a computer program, DYNARACK, for dynamic analysis to demonstrate the structural adequacy of the WBN spent fuel rack design under the earthquake loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor of the storage pool. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements, as defined in the program, was used to simulate three dimensional dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Two model analyses were performed: the 3-D single rack model analysis and the 3-D whole pool multi-rack (WPMR) analysis. For the 3-D single rack model analysis, three rack geometries were considered for the calculation of stresses and displacements: (1) 6.1 ft (W) x 7.8 ft (L) x 12.5 ft (H) and (2) 13 ft (W) x 13 ft (L) x 12.5 ft (H), where W, L and H are defined as width, length and height of a rack, respectively. Each rack was considered fully loaded and half loaded with two different coefficients of friction between the rack and the pool floor ($\mu$=0.2 and 0.8) to identify the worst case response for rack movement and for rack member stresses and strains. In the WPMR model analysis, 34 free standing racks were considered to investigate the fluid-structure interaction effects between racks and pool walls as well as those among the racks.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration time histories) were generated from the design response spectra defined in the FSAR. TVA demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density function compatible with the design response spectra as discussed in SRP Section 3.7.1.

A total of 12 3-D single rack model analyses was performed. The results of the analyses show that the maximum displacements of the racks at the top and the baseplate corners are 0.7 inch and 0.05 inch, respectively, assuring that there are no rack-to-wall or rack-to-rack impacts under the load combinations (Level A, B and D service limits). The analyses results indicate that there are large safety margins against overturning of the racks as evidenced by the small rack movements and, thereby, the structural integrity and stability of the racks and fuel assemblies are maintained. In addition, the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (1986 edition), Section III, Subsection NF, as shown in Table 2.1 of the February 24, 1997 submittal. The results show that all induced stresses under the load combinations (Level A, B and D service limits) are smaller than the corresponding allowable stresses specified in the ASME

Boiler and Pressure Vessel Code, Section III, Subsection NF, indicating that the rack design is adequate.

In the 3-D WPMR analyses, 34 fully loaded racks were considered and were subjected to the dynamic loading conditions (Level D service limit). The results of the multi-rack analysis indicate that the calculated stresses on a rack are higher than those obtained from the single rack analyses. However, all calculated stresses for the WPMR analyses are smaller than the corresponding allowable stresses of the ASME Code. In addition, the results show that there are no rack-to-wall or rack-to-rack impacts as the result of an SSE; assuring that the structural integrity and stability of the racks are maintained.

TVA calculated the weld stresses of the rack at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal and cell-to-cell connections) under the dynamic loading conditions. TVA demonstrated that all the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code Section III, Subsection NF, indicating that the weld connection design of the rack is adequate.

In addition, TVA analyzed a single cell for shear buckling. TVA calculated the factors of safety of 1.22 and 12.51 for the flange and web of a cell, respectively, under the maximum compression stress due to an SSE and demonstrated that there is no shear buckling in a cell.

Based on: (1) the TVA's comprehensive parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack), (2) large factor of safety of the induced stresses of the rack when they are compared to the corresponding allowables provided in the ASME Boiler and Pressure Vessel Code, Section III, (3) a reasonable assurance that there is no rack-to-wall and rack-to-rack impacts, and (4) TVA's overall structural integrity conclusions supported by both single analyses and multi-rack analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

It is quite likely that the racks will move during seismic events. Accordingly, the staff concluded that TVA should inspect the gaps between racks after the occurrence of an earthquake equivalent to or larger than an operating basis earthquake to insure the gaps remain. TVA responded to this concern in its submittals dated February 24 and July 15, 1997 by confirming that the rack gaps are specifically identified in the abnormal operating instruction as one of the plant features of the inspection following an earthquake. This is acceptable.

The staff concluded that TVA should inspect for structural degradation, particularly of the welds of the rack cells, of the PaR racks which were previously used at SQN. In its response of June 26, 1997, TVA committed to perform weld examinations of the accessible welds which attach the spent fuel storage cells to the top and bottom castings of the racks. TVA's submittal of July 15, 1997 indicated that this weld examination had been completed. TVA stated that the welds were found to conform to the weld detail shown on the manufacturer's drawing and were described by the examiners to be of good quality. TVA also stated that the examinations did not identify any indications which would cause concern about the structural integrity of the welds. The staff concludes that TVA's stated findings resolve this concern.

2.3.3 Spent Fuel Storage Pool

The spent fuel pool structure is a reinforced concrete structure and is designed as a seismic Category I structure. The dimensions of the WBN pool structure are approximately 31.8 feet wide, 53.5 feet long and 48 feet deep; the reinforced concrete is 7 feet thick. The internal surface of the pool structure is lined with stainless steel to ensure water tight integrity.

The pool structure was analyzed by using the finite element computer program, ANSYS, to demonstrate the adequacy of the pool structure under fully loaded fuel racks with all storage locations occupied by fuel assemblies. The fully loaded pool structure was subjected to the load combinations specified in the WBN FSAR.

Table 8.4.1 of the licensee's October 23, 1996 submittal shows the predicted factors of safety varying from 1.05 to 1.23 for bending moments of the concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the TVA pool structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading and SSE loading conditions. Thus, the storage fuel pool design is acceptable.

2.3.4 Fuel Handling Accident

The following three refueling accident cases were evaluated by TVA: (1) two cases for drop of a fuel assembly with its handling tool, which impacts the baseplate (deep drop scenario) and (2) one case for drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario).

The analysis results of Accident Case (1) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area; therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis results of Accident Drop Case (2) show that damage will be restricted to a depth of 3.06 inches below the top of the rack, which is above the active fuel region. The staff reviewed TVA's analysis results in its submittal of October 23, 1996 and concurs with its findings. This is acceptable based on the TVA's structural integrity conclusions supported by the parametric studies.

2.3.5 Structural Engineering Conclusion

Based on the review and evaluation of the TVA's October 23, 1996 submittal, and additional information and analysis provided by TVA in its February 24, May 28, June 26 and July 15, 1997 submittals, the staff concludes that TVA's structural analysis and design of the spent fuel rack modules and the spent fuel pool structure are adequate to withstand the effects of the required loads. The analysis and design are in compliance with current licensing basis set forth in the FSAR and applicable provisions of the SRP and, therefore, are acceptable.

## 2.4 Materials Engineering

### 2.4.1 Structural Materials

The following structural materials are used in the spent fuel racks manufactured by PaR:

- Poison can and outer tubes: 304 stainless steel, ASTM Standard A-666-72 Grade B
- Top and bottom grid casting: CF-3M, ASTM Standard A-296-77
- Threaded pedestal foot: 17-4 PH, ASTM Standard A-564-66

The following structural materials are used in the spent fuel racks to be provided by Holtec International:

- Storage cell structures and internally threaded support legs: 304L stainless steel, ASME Standard SA240
- Externally threaded support spindle: precipitation hardened stainless steel ASME Standard SA564-630
- Weld material: ASME Standard SFA 5.9 ER308L

All these materials have been previously used in many applications. They were exposed to environments similar to or more severe than those existing in the spent fuel pool of WBN without experiencing any observable corrosion damage. In addition, the racks made by PaR were exposed to the spent fuel pool environment for 13 years at the SQN plant and did not indicate signs of corrosion.

### 2.4.2 Poison Material

In both types of spent fuel racks, manufactured by PaR and by Holtec, Boral is utilized as a neutron absorbing material. Boral is a cermet composite material made of Type 1100 aluminum and boron. The composite panel consists of a plate of boron carbide in a Type 1100 aluminum matrix clad in Type 1100 aluminum sheets. The Type 1100 aluminum material imparts sufficient pitting and general corrosion resistance by forming an aluminum oxide layer on its surface when exposed to oxidizing environments. This oxide is stable in an environment with a pH range of 4.5 to 8.5. The boron carbide particles in Boral panels have been shown to have good structural compatibility with the Type 1100 matrix material. Despite these corrosion preventing properties of Boral, some corrosion is expected. Although this will not result in a significant depletion of boron and resulting degradation of its neutron absorbing properties, some generation of hydrogen from corrosion of aluminum can occur when Boral is exposed to the spent fuel pool water. This effect is more pronounced in new panels which do not yet have well formed protective oxide film. This hydrogen, when not vented, could cause swelling of the Boral wrappers and resultant deformation of storage cells. In order to prevent this from occurring, the new racks, manufactured by Holtec, will have vented Boral wrappers. The spent fuel racks, manufactured by PaR, have Boral seal welded in wrappers. This has proved to be an effective corrosion preventive method, and these racks did not exhibit swelling during their 13-year service in SQN. Due to these considerations, the licensee does not propose to have a specific

surveillance program for Boral containing racks. However, after installation, the racks will be subjected to a 100-percent drag test which will check for storage cell distortion caused by hydrogen-induced swelling. Also, at each refueling, drag will be monitored during insertion and withdrawal of fuel assembles. In addition, coupons made of an identical Boral material have been installed in the Browns Ferry plant and they will provide some indication as to the stability of Boral. These measures are deemed by the licensee to be sufficient to ensure that information about the condition of Boral could be obtained without recourse to a specific surveillance program. The staff reviewed the proposed program and believes that it should provide sufficient indication of the condition of Boral in the spent fuel racks.

### 2.4.3 Conclusions

Based on its evaluation, the staff finds that the materials in the spent fuel racks, manufactured by Programmed and Remote System Corporation and the new rack designed and manufactured by Holtec International are compatible with the environment of the WBN spent fuel pool and will not undergo material degradation which could affect the ability of the storage racks to safely store spent fuel. In the racks designed by Holtec, Boral may exhibit slight corrosion before a protective film oxide on aluminum cladding is formed. However, the resulting generation of hydrogen will not cause damage to the racks because a provision exists for venting this hydrogen from the Boral wrappers. The licensee does not intend to establish any specific surveillance program for Boral. However, drag tests, to be performed at each refueling outage, will ensure that no distortion of fuel cells has occurred, and information on the behavior of Boral will be available from the Boral surveillance coupons installed in the Browns Ferry plant.

## 2.5 Radiological Protection

### 2.5.1 Occupational Dose Control

The operations involved in the reracking will utilize detailed procedures that fully consider as low as reasonably achievable (ALARA) principles. The licensee has performed similar operations in the past at other plants. The licensee believes that the reracking can be safely and efficiently accomplished at the WBN plant with minimum radiation exposure to personnel.

There will be no spent fuel or other contaminated material in the WBN spent fuel storage pool when the PaR racks transferred from the SQN are installed. These storage racks were previously in service at SQN and are contaminated. The additional "baby" Holtec designed racks for the periphery of the SFP will not be installed in the 1997 WBN rerack activity, but will be utilized at a later date when the fuel pool inventory, or other factors, warrant their deployment. Future installation of these racks will be done remotely while spent fuel is stored in the pool. These remote tools include long handled tools for leveling the racks, remotely engageable and disengageable lift fixtures for old rack removal and new rack installation. The operations involved in reracking will utilize detailed procedures prepared with full consideration of ALARA principles.

The licensee estimated that the total occupational dose for the planned reracking activities will be 2.5 person-rem, including any necessary diving operations. The licensee's operating experience has shown that area radiation dose rates from radionuclides in the pool water are generally 1 to 5 mrem/hr.

Dose rates on the pool bridge crane platform are 4 to 5 mrem/hr. This overall estimate is a sum of the individual dose estimates for each of the anticipated activities to be performed during the reracking operation. The licensee will use divers only as a contingency. If divers are used, the licensee has committed to the guidance provided in Appendix A ("Procedures for Diving Operations in High and Very High Radiation Areas") to RG 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Power Plants."

Normal operational radiation levels in zones surrounding the pool are not expected to be significantly affected. Existing shielding around the pool (such as the depth of the water and concrete walls) will provide additional protection. When the peripheral "baby" uncontaminated racks are added, the concentrations of radionuclides in the pool may increase due to crud deposit spalling from spent fuel assemblies which are shuffled. Dose rates due to crud spalling will be detected by the refuel floor radiation monitors located at the northeast and southwest corners of the pool. These monitors readout and alarm in the main control room. Additionally, RADCON technicians would be present during this fuel shuffling operation. The spent fuel pool cooling and cleaning system would further reduce the dose rates to acceptable ALARA levels.

The licensee notes that detailed procedures that fully consider the ALARA principle will be utilized. In addition, the licensee stated that continuous air samplers will be utilized when a potential for significant airborne activity exists and that personnel will wear protective clothing and, as appropriate, respiratory protection equipment. Further, work activities are to be governed by Radiation Work Permits that specify appropriate radiation protection measures. In addition to the routine use of self-reading dosimeters and thermoluminescence dosimeters, extremity badges and alarming dosimeters will be used, as appropriate. The licensee further states that work activities, personnel traffic, and equipment movement will be monitored and controlled to minimize contamination and to ensure that exposures are maintained ALARA.

On the basis of its review of the licensee's application, the staff finds the proposed radiation protection aspects of the reracking of the spent fuel pool acceptable.

### 2.5.2 Solid Radioactive Waste

The licensee stated, in section 9.2 of its October 23, 1996 submittal, that no significant increase in the volume of solid radioactive wastes is expected with the expanded storage capacity. The originally installed racks will be moved from the WBN plant before any spent fuel has been stored in them. They are not contaminated and will not contribute to the volume of solid radioactive waste. Shipping containers will meet the requirements of NRC and Department of Transportation regulations pertaining to radioactive waste shipments, including limitations with respect to the waste surface dose and radionuclide activity distribution.

On the basis of its review, the staff finds that the licensee's plan for handling and disposing of solid radioactive waste generated in connection with the planned reracking operation meets regulatory requirements and is, therefore, acceptable.

2.5.3 Design Basis Accidents (DBA)

In its application, the licensee evaluated the possible consequences of postulated accidents and described the means for mitigating these consequences should they occur. This evaluation included spent fuel handling accidents. On the basis of its analysis, the licensee concluded that the effects of the proposed TS changes are small and that the calculated consequences are within regulatory requirements and staff guideline dose values. By letter dated April 4, 1997, the licensee revised the dose calculations based on the requirements of NUREG/CR-5009, Paragraph 2, which states that when compared with Regulatory Guide 1.25 requirements, the release fractions for I-131 are higher (12%) for extended burnup and higher enrichment fuel. The licensee's analysis calculated the doses for the 0-2 hour at the exclusion area boundary to be 2.18 rem to the thyroid and 0.51 rem to the thyroid for the low population zone. These doses are within the SRP (NUREG-0800) acceptance criteria of 75 rem to the thyroid.

In the Safety Evaluation for Watts Bar, NUREG-0847, Supplement No. 15, dated June 1995, the staff estimated the offsite doses from a release of radionuclides to the atmosphere from a fuel handling accident. The staff concluded that the plant's mitigative features would reduce the doses from this DBA to below the doses specified in SRP (NUREG-0800), Section 15.7.4.

The staff evaluated the consequences of operation at a bounding value (60,000 MWD/MT) because of the licensee's reference to the use of more highly enriched fuel (up to 5.0 weight percent U-235). In the table below, the fuel handling accident doses associated with extended burnup and higher enrichment fuel, as well as that contained in the current licensing basis, are presented and compared to the guideline doses in SRP (NUREG-0800) Section 15.7.4.

EXCLUSION AREA BOUNDARY THYROID DOSES FROM WBN FUEL HANDLING ACCIDENT (REM)

| | |
|---|---|
| Staff Evaluation NUREG-0847, SSER-15 dated June 1995 | 1.5 rem |
| Bounding Estimate for Extended Burnup fuel[3] (60,000 MWD/MT) | 1.8 rem |
| Acceptance Criteria (NUREG-0800 Section 15.7.4) | 75.0 rem |

The staff concludes that the only potential increased radiological consequences resulting from a reracking with extended burnup and higher fuel enrichment are the thyroid doses; these doses remain well within the acceptance criteria given in NUREG-0800 and are, therefore, acceptable.

[3] According to NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors, PNL, 1987," increasing fuel enrichment to 5.0 wt% U-235 with a maximum burnup of 60,000 MWD/MT increases the doses from radioiodine for fuel handling accident by a factor of 1.2.

## 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. 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 2, 1997 (62 FR 15733). In this finding, the Commission determined that issuance of these amendments would 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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