March 22, 1999

Mr. Otto L. Maynard
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. MA1294)

Dear Mr. Maynard:

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated March 20, 1998, as supplemented by letters dated May 28, June 30, August 28, September 4, November 20, and December 8, 1998.

The amendment revises the TS to support a modification to the plant to increase the storage capacity of the spent fuel pool and increase the nominal fuel enrichment to 5% weight percent of U-235. The amendment also revises the TS to allow the storage of an additional 279 assemblies in the cask loading pit.

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

Sincerely,
Original Signed By
Kristine M. Thomas, Project Manager
Project Directorate IV-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Amendment No. 120 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

*For previous concurrences see attached ORC
Document Name: WCMA1294.WPD

DISTRIBUTION

Docket File	OGC, 015B18
PUBLIC	KBrockman, RIV
PDIV-2 Reading	GHill (2)
JZwolinski/SBlack	WBeckner, O11E22
WBateman	LYandell, RIV
KThomas	ACRS
EPeyton	CNorswort
LHurley, RIV	JKilcrease, RIV

OFC	PDIV-2	PDIV-2	PERB*	SRXB*	SPLB*
NAME	KThomas	EPeyton	TEssig	JWermiel	TMarsh
DATE	3 / 22 /99	3 / 22 /99	3 / 5 /99	3 / 3 /99	3 / 16 /99

OFC	EMCB*	ECGB*	OGC*
NAME	TSullivan	GBagchi	STurk
DATE	3 / 18 /99	3 / 13 /99	3 / 16 /99

OFFICIAL RECORD COPY

cc w/encls:
Jay Silberg, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW
Washington, D.C. 20037

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 311
Burlington, Kansas 66839

Chief Engineer
Utilities Division
Kansas Corporation Commission
1500 SW Arrowhead Road
Topeka, Kansas 66604-4027

Office of the Governor
State of Kansas
Topeka, Kansas 66612

Attorney General
Judicial Center
301 S.W. 10th
2nd Floor
Topeka, Kansas 66612

County Clerk
Coffey County Courthouse
Burlington, Kansas 66839

Vick L. Cooper, Chief
Radiation Control Program
Kansas Department of Health
and Environment
Bureau of Air and Radiation
Forbes Field Building 283
Topeka, Kansas 66620

Chief Operating Officer
Wolf Creek Nuclear Operating Corporation
P. O. Box 411
Burlington, Kansas 66839

Supervisor Licensing
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
8201 NRC Road
Steedman, Missouri 65077-1032

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. NPF-42

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

   A. The application for amendment to the Wolf Creek Generating Station (the facility) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation) dated March 20, 1998, as supplemented by letters dated May 28, June 30, August 28, September 4, November 20, and December 8, 1998, 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, as amended, 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 license 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-42 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 120 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance to be fully implemented no later than December 31, 1999, except that the racks in the cask loading pit may be installed at a future time after the completion of the next refueling outage. While the spent fuel pool reracking modification is in progress, both the technical specifications issued through Amendment No. 119, and those technical specifications being amended by Amendment No. 120 will be applicable.

FOR THE NUCLEAR REGULATORY COMMISSION

Kristine M Thomas

Kristine M. Thomas, Project Manager
Project Directorate IV-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 22, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 9-15	3/4 9-15
3/4 9-16	3/4 9-16
B 3/4 9-3	B 3/4 9-3
5-7	5-7

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12 Spent fuel assemblies stored in either the Spent Fuel Pool or cask loading pit shall be subject to one of the following conditions:

a. Spent fuel assemblies regardless of burnup can be placed in Region 1.

b. Spent fuel assemblies within the burnup-enrichment range shown in Figure 3.9-1 for Region 2 fuel can be placed in Region 2 or 3.

c. Spent fuel assemblies within the burnup-enrichment range shown in Figure 3.9-1 for Region 3 fuel shall be placed in Region 3.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool or cask loading pit.

ACTION:

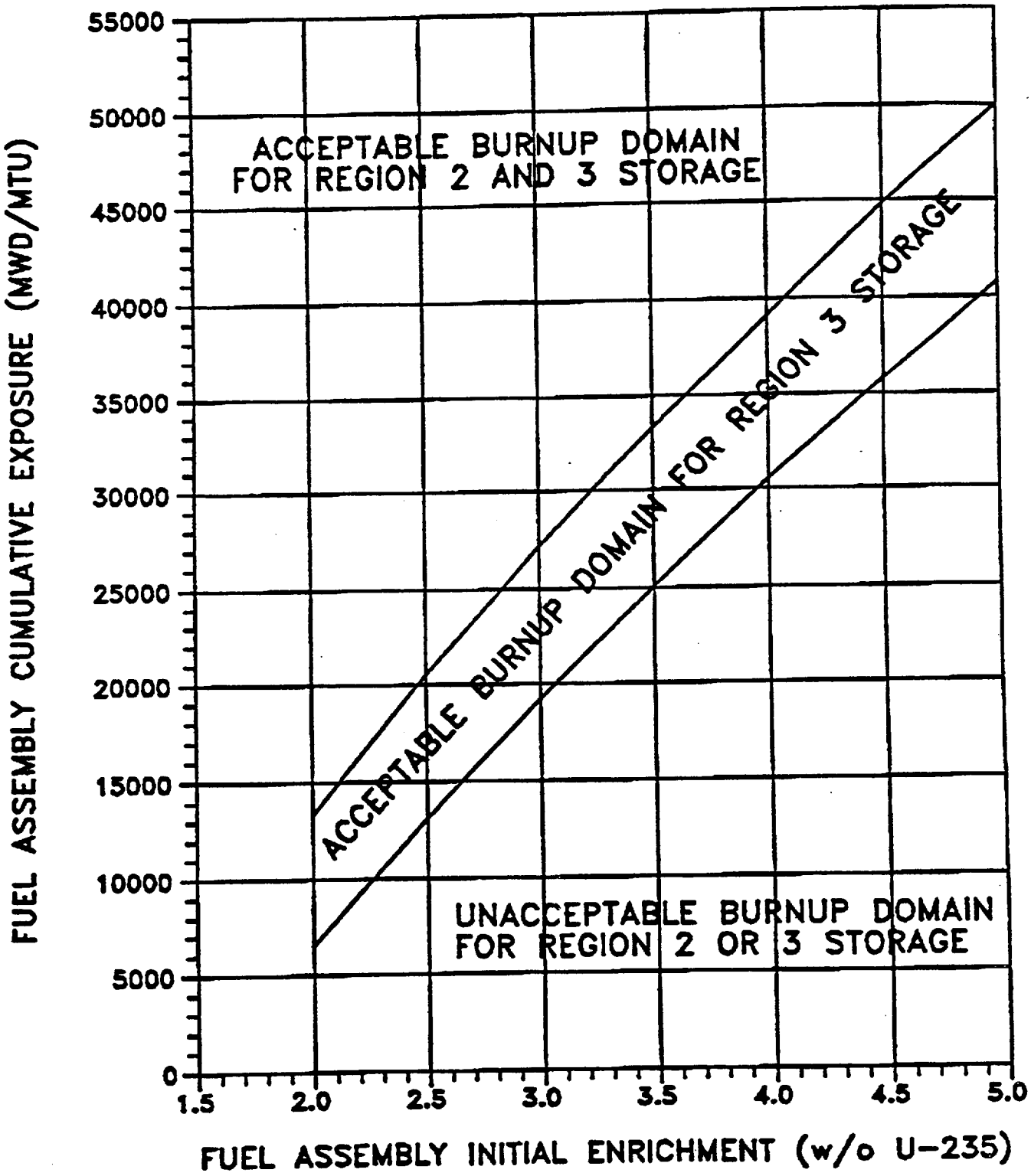
a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move non-complying fuel assemblies to Region 1. Until the requirements of the above specification are satisfed boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The burnup of each spent fuel assembly stored in Regions 2 or 3 shall be ascertained by analysis of its burnup history and independently verified prior to storage in Regions 2 or 3. A complete record of such analysis shall be kept for the time period that the spent fuel assembly remains in Regions 2 or 3 of the spent fuel pool or cask loading pit.

FIGURE 3.9-1
MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGIONS 2 AND 3



3/4.9.9 CONTAINMENT VENTILATION SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

The racks for the Spent Fuel Pool and cask loading pit are designed for storage of both new fuel and irradiated fuel. Prior to storage of fuel assemblies in either the Spent Fuel Pool or cask loading pit, overall pool storage Regions shall be prepared in accordance with administrative controls. The restrictions placed on fuel assemblies stored in the spent fuel pool and cask loading pit ensure inadvertent criticality will not occur. Region 1 is designed to accommodate new fuel with a maximum nominal initial enrichment of 5.0 weight percent U-235, or spent fuel regardless of the discharge fuel burnup. Region 2 and Region 3 are designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.9-1, in the accompanying LCO.

3/4.9.13 EMERGENCY EXHAUST SYSTEM - FUEL BUILDING

The limitations on the Emergency Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1975 and N510-1980 will be used as procedural guides for surveillance testing.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC approved codes and methods and shown by test or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rod assemblies shall be hafnium, silver-indium-cadmium, or a mixture of both types. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

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

b. For a pressure of 2485 psig, and

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

VOLUME

5.4.2 The total volume of the Reactor Coolant System, including pressurizer and surge line, is 12,135 ± 100 cubic feet at a nominal T_{avg} of 557°F.

5.5 METEOROLOGICAL TOWER LOCATION

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

5.6 FUEL STORAGE

CRITICALITY

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

a. With a k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the USAR. This is based on new fuel with maximum nominal enrichment of 5.0 weight percent U-235 in Region 1 and on spent fuel with combination of initial enrichment and discharge exposures, shown in Figure 3.9-1, for Regions 2 and 3, and

b. With a nominal 8.99 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. For fuel with nominal enrichments greater than 4.6 weight percent U-235, the combination of enrichment and integral fuel burnable absorbers shall be sufficient so that the requirements of Specification 5.6.1.1.a are met. Integral fuel burnable absorbers are not required for Region 1 locations on the periphery of the pool, adjacent to a concrete wall.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 2040 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2363 fuel assemblies. The cask loading pit is designed and shall be maintained with a storage capacity limited to no more than 279 assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By letter dated dated March 20, 1998, as supplemented by letters dated May 28, June 30, August 28, September 4, November 20, and December 8, 1998, Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station. The proposed changes would revise the technical specifications to support a modification to the plant to increase the storage capacity of the spent fuel pool and increase the nominal fuel enrichment to 5% weight percent (w/o) of U-235.

2.0 BACKGROUND

As discussed in its submittals, the licensee is planning to rerack the spent fuel pool and to add fuel storage racks to the cask loading pit to be able to accommodate a full-core discharge through the end of its license period which ends in 2025. The current spent fuel pool racks will be replaced by higher density racks and also sparger lines in the pool will be truncated to maximize storage space. In addition, at a later date, racks will be placed within the cask loading pit.

The expansion effort will result in an increase in the total storage space from 1340 to 2642 fuel assemblies, 279 of which will be located in the cask loading pit and the remainder in the pool itself. The new racks provide a closer assembly-to-assembly spacing to increase the overall capacity. The racks also contain Boral for active neutron absorption. These racks also accommodate fuel enriched as much as 5% nominal weight percent U-235 in a mixed zone three region storage configuration.

The supplemental letters dated June 30, August 28, September 4, November 20, and December 8, 1998, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the FEDERAL REGISTER on July 13, 1998 (63 FR 37601).

3.0 EVALUATION

3.1 Criticality

The licensee's analysis of the reactivity effects of fuel storage in the spent fuel racks was performed with the three-dimensional Monte Carlo code, KENO5a, using the 238-group SCALE cross-section library. Independent verification calculations were performed with MCNP4a, a continuous energy three-dimensional code. Since the KENO5a code package does not have burnup capability, depletion analyses were made with CASMO-3, a two-dimensional multigroup transport theory code. The determination of small reactivity increments due to manufacturing tolerances was also made with CASMO-3. These codes are widely used for the analysis of spent fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the Wolf Creek Generating Station (WCGS) spent fuel racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and B-10 loading in the absorber. The two independent methods of analysis (KENO5a and MCNP4a) showed good agreement both with experiments and with each other. The comparison between different analytical methods is an acceptable technique for validating calculation methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO5a calculations, a minimum of 5,000,000 neutron histories in 1,000 generations of 5,000 neutrons each were accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO5a reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the WCGS storage racks with a high degree of confidence.

The spent fuel storage racks are normally flooded with water borated to at least 2000 parts per million (ppm) of boron, which results in a large subcriticality margin under actual operating conditions. NRC Standard Review Plan (SRP) Section 9.1.2 provides criteria that are to be satisfied to assure criticality safety under normal and accident conditions, and to ensure conformance to the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 62 (GDC 62) for the prevention of criticality in fuel storage and handling. This criterion states that the maximum reactivity (k_{eff}) of the racks containing fuel of the highest anticipated reactivity and flooded with unborated water must not exceed 0.95. The maximum calculated reactivity must include a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true k_{eff} will not exceed 0.95 at a 95% probability, 95% confidence (95/95) level.

The licensee's criticality analyses were performed with several conservative assumptions which tend to maximize the rack reactivity. These include:

(1) unborated pool water at the temperature yielding the highest reactivity (4°C) over the expected range of water temperatures,

(2) an infinite array of storage cells in the lateral direction (except for the assessment of peripheral effects and certain abnormal conditions where neutron leakage is inherent), and

(3) neutron absorption effect of minor structural material is neglected.

Since the WCGS contains Westinghouse Standard (STD) Vantage-5H (V5H) and Optimized (OFA) fuel designs, the licensee performed calculations for each of these fuel types. The OFA fuel exhibited the highest reactivity at zero burnup. Therefore, the design basis fuel assembly was a Westinghouse OFA with a 17x17 array of fuel rods containing uranium oxide (UO2) at a maximum initial enrichment of 5.0 weight percent (w/o) U-235, with 25 fuel rods replaced by 24 control rod guide tubes and one instrument thimble. At higher burnups, the V5H assembly type is the more reactive and, therefore, for the burnup dependent Regions 2 and 3, reactivities equivalent to the V5H assembly were used.

The staff concludes that the licensee made appropriate conservative assumptions.

Spent fuel storage is designated into regions based on administrative controls. Region 1 is designed to accommodate new (fresh) fuel with a maximum nominal enrichment of 4.6 w/o U-235. To enable the storage of fuel assemblies with nominal enrichments greater than 4.6 w/o U-235, the licensee utilized the concept of reactivity equivalencing. In this technique, which has been previously approved by the NRC, credit is taken for the reactivity decrease due to the integral fuel burnable absorber (IFBA) material coated on the outside of the UO2 pellet. Region 2 and Region 3 are designed to accommodate fuel of up to 5.0 w/o U-235 initial enrichments which has accumulated minimum irradiation levels within the acceptable burnup domain depicted in the proposed TS Figure 3.9-1.

The licensee proposes to use a "Mixed-Zone Three-Region" (MZTR) configuration and/or a checkerboard configuration for storage of fuel assemblies in the spent fuel pool. In a MZTR configuration, Region 1 cells are only located along the outside periphery of the rack modules and must be separated by one or more Region 2 storage cells. Region 1 storage cells may be located directly across from one another when separated by a water gap. The outer rows of alternating Region 1 and 2 storage cells must be further separated from the internal Region 3 storage cells by one or more Region 2 cells. Fresh assemblies with enrichment greater than 4.6 w/o U-235 and without IFBA rods must be stored in any peripheral Region 1 storage cell that is next to a concrete wall. In the checkerboard configuration, assemblies are placed in an alternating checkerboard-style pattern with empty cells (i.e., assemblies are surrounded on all four sides by empty cells except at the checkerboard boundary).

For the analysis of the nominal MZTR storage configuration, the analysis accounted for uncertainties due to boron loading tolerances, Boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, fuel enrichment and density tolerances, water-gap spacing between modules, and eccentric fuel positioning. These uncertainties were appropriately determined at the 95/95 probability/confidence level. In addition, a calculational bias and uncertainty were determined from benchmark calculations as well as an allowance for uncertainty in depletion calculations and the effect of the axial distribution in burnup. The final maximum calculated reactivity resulted in a k_{eff} of 0.9428 for 4.6 w/o U-235 fuel with no IFBA rods in Region 1 and 0.9400 for 5.0 w/o U-235 fuel with 16 IFBA rods in Region 1, when combined with all known uncertainties. Fuel assemblies in Regions 2 and 3 were assumed to meet the minimum required burnup as a function of nominal initial enrichment given in the proposed TS Figure 3.9-1.

Similar calculations for the checkerboard storage configuration resulted in a 95/95 keff of 0.8551 for 5.0 w/o U-235 fuel with 16 IFBA rods and 0.8602 for 4.6 w/o fuel with no IFBA rods.

These values meet the staff's criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level. Therefore, the staff finds the proposed MZTR and checkerboard storage configurations to be acceptable.

Proposed TS Figure 3.9-1 shows that Regions 2 or 3 can safely accommodate fuel of various initial enrichments up to 5.0 w/o U-235 and discharge burnups provided the combination falls within the acceptable domain illustrated by the upper line. Assemblies meeting the burnup requirements of the lower line may only be stored in Regions 1 or 3. This reactivity equivalencing method is the standard method used for storage rack reactivity evaluations and is acceptable.

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 area in the proposed TS Figure 3.9-1, or dropping an assembly between the rack modules and the concrete wall of the pool, 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 boron in the pool water, 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 2000 ppm of boron in the pool is required by plant chemistry procedures and verified weekly. The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. In fact, the licensee has confirmed that a minimum boron concentration of only 500 ppm boron would be adequate to assure that the limiting k_{eff} of 0.95 is not exceeded for the worst accident.

The licensee has requested the following technical specification changes to support the proposed spent fuel pool reracking modification. Based on the above evaluation, the staff finds these changes acceptable, as well as the associated TS Bases changes.

1. TS 3.9.12 has been revised to reflect the conditions required for fuel assembly storage in Regions 1, 2, and 3. The associated TS Figure 3.9-1, which shows the minimum required fuel assembly burnup as a function of initial enrichment for allowed storage in Regions 2 and 3, has been revised.

2. TS 5.6.1.1a has been modified to allow spent fuel storage of assemblies with maximum nominal enrichment of 5.0 w/o U-235 in Region 1 and storage in Regions 2 and 3 based on acceptable combinations of initial enrichment and discharge exposure as shown in Figure 3.9-1.

3. TS 5.6.1.1b has been modified to reflect the reduced center-to-center distance between fuel assemblies from 9.24 to 8.99 inches for the new fuel storage racks.

4. TS 5.6.1.1c has been created to insert the requirement for sufficient IFBAs for storage of fuel with nominal enrichments greater than 4.60 w/o U-235.

5. TS 5.6.3 has been modified to reflect the increased fuel pool storage capacity of the spent fuel storage pool to 2363 fuel assemblies and of the cask loading pit to 279 fuel assemblies.

Conclusion

Based on the review described above, the staff finds that the criticality aspects of the proposed modifications to the WCGS spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Although the WCGS 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 by the licensee 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.

3.2 Plant Systems

3.2.1 Spent Fuel Pool Cooling and Cleanup System

The WCGS spent fuel pool cooling and cleanup system (SFPCCS) consists of two cooling trains, a cleanup loop, and a surface skimmer loop. The primary safety function of the SFPCCS is to adequately transport the SFP heat load to the component cooling water system and thereby maintain the bulk pool temperature within specified limits. The cooling portion of the system consists of two 100 percent capacity, seismically qualified, safety-related cooling trains that remove decay heat generated by irradiated fuel stored in the SFP and the cask loading pit. Each train consists of a horizontal centrifugal pump, shell and U-tube heat exchanger, strainer, manual valves, and the instrumentation required for system operation. The non-seismic reactor makeup water tank provides the normal source of makeup water to the SFP via the reactor makeup water pumps. A second source of makeup water to the SFP is provided by the refueling water storage tank via the non-seismic SFP cleanup pumps and piping. The safety-related and seismically-qualified makeup source to the SFP is provided by the essential service water (ESW) system via the ESW pumps. The cleanup portion of the SFPCCS purifies the water in the SFP, cask loading pit, transfer canal, refueling pool, and the refueling water storage tank.

3.2.2 Decay Heat Load Limit

The licensee performed an evaluation of the maximum spent fuel decay heat load for the SFP in order to control the pool temperature within a specified limit. The SFP heat loads will be maintained below these specified limits by performing refueling outage specific evaluations of the complete pool decay heat load. The licensee will require these outage specific calculations through administrative controls prior to each discharge of spent fuel into the pool. The heat load calculations will be developed in accordance with NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling," or ANSI/ANS 5.1-1979. The calculations will take into account all previously discharged fuel and the predicted heat load for each newly discharged fuel batch.

The following conservatisms will be included in the licensee's decay heat load limit calculations:

1. SFPCCS heat exchanger thermal performance will be based on the design maximum fouling and plugging level. This will conservatively minimize the heat rejection capability of the SFPCCS.

2. Thermal inertia induced transient effects resulting in a lag in bulk pool temperature response will be neglected. This will conservatively lower the calculated decay heat load limit by forcing the peak decay heat load to coincide with the peak pool temperature.

3. In calculating the SFP evaporation heat losses, the SFP building will be assumed to have the maximum ambient air temperature of 110°F and 100 percent relative humidity to minimize the credit for evaporative heat loss.

To develop the maximum decay heat loads, the licensee evaluated the following three scenarios (1) partial core offload, (2) full core offload, and (3) SFP post-LOCA scenario.

The partial core offload scenario assumes that approximately one half of the fuel assemblies are offloaded. It further assumes the fuel assemblies are discharged 100 hours after reactor shutdown and includes the additional heat load from fuel assemblies already stored in the pool. The licensee determined the maximum pool heat load to be 27.15 MBtu/hr, which would result in a steady state bulk pool temperature limit of 140°F, assuming a single failure in one train of the SFP cooling system.

A full core off-load is the general practice for planned refueling outages at WCGS. Currently, the WCGS UFSAR describes a SFP bulk pool temperature limit of 160°F, assuming a single failure in one train of the SFP cooling system. In the proposed full core off-load scenario, the maximum pool heat load would be based on the decay heat associated with a full core (193 assemblies) removed from the reactor following a reactor shutdown, plus the additional heat load from assemblies already stored in the SFP. The licensee determined the maximum pool heat load to be 63.41 MBtu/hr, which would result in a steady state bulk pool temperature limit of 170°F, assuming a single failure in one train of the SFP cooling system. Makeup water would normally be provided by the reactor makeup water tank via the reactor makeup water pumps; however, the other sources of makeup water would be available, if necessary.

The staff determined that the maximum bulk SFP temperature of 170°F for the full core offload scenario exceeds the guidance in American Concrete Institute (ACI) Standard 349. ACI 349 states in part "for normal operation or any other long-term period, the SFP temperatures shall not exceed 150 degrees." The licensee stated that, although a full core offload is the normal practice during refueling outages, the associated decay heat load is not considered to be a long term normal operation heat load. The licensee determined that the maximum time period that the pool temperature would be above 150°F would be less than 9 days per fuel cycle and this temperature would be reached in only in a cross-sectional portion of the SFP concrete walls.

The staff questioned the impact of increasing the peak SFP temperature to 170°F on the structural analyses completed for the SFP and cask loading pit. The licensee provided information demonstrating that the structural evaluations included conservatisms that compensated for any potential effects that may result from the increased peak SFP

temperature. These conservatisms are described in Section 3.4.2 of this safety evaluation. The staff has determined that the conservatisms in the structural analyses are adequate to ensure the proposed increase in peak SFP temperature would not alter the structural analyses results. Therefore, as described in Section 3.4.2 of this safety evaluation, the staff finds the increase in peak SFP temperature to be acceptable.

In the post-LOCA scenario, the heat load on the SFP is assumed to be equivalent to that assumed for the partial core discharge scenario. To maximize pool heat, the licensee assumed that the core is refueled and restarted within 100 hours of shutdown, and the resultant decay heat load on the SFP is that of approximately half the core plus the previous fuel discharges. It is also assumed that all cooling is suspended in the SFP cooling system, while component cooling water is diverted to the residual heat removal system. At four hours post-LOCA, component cooling water flow is re-established to the SFP cooling system at 50 percent of its capacity. The licensee determined that throughout and subsequent to this SFP cooling transient, the surface temperature of the SFP is limited to less than boiling. The staff finds this to be acceptable since it meets the recommendation for the abnormal maximum heat load limit of maintaining the SFP temperature to less than boiling as stated in SRP 9.1.3. Makeup water would be provided by the ESW system via the ESW pumps, if the two non-seismic makeup sources were unavailable.

As discussed in Section 2.0 of this safety evaluation, the licensee's application includes the installation of three high density racks in the cask loading pit. The staff questioned how sufficient cooling would be provided to the fuel located in the cask loading pit and what controls would ensure that newly off-loaded fuel would not be placed in the cask loading pit. The licensee indicated that the cask loading pit would be cooled by the passive, buoyancy-driven exchange of water with the SFP. Cooling was demonstrated by a three-dimensional computational fluids model, FLUENT 3D, which included the cask loading pit and its interconnecting slot that is located between the SFP and the cask loading pit. The results showed that the temperature contours through the interconnecting slot had no substantial difference between the bulk pool temperature and the cask loading pit temperatures. The staff reviewed the licensee's results and found them to be acceptable. As documented in a letter from the licensee dated August 28, 1998, the licensee committed to use administrative controls in the form of plant procedures to ensure that newly off-loaded fuel is not stored in the cask loading pit consistent with the modeling.

Conclusion

Based on the staff's evaluation and the guidance in SRP 9.1.3, the staff concludes that the thermal-hydraulic aspects of the proposal for increasing the WCGS spent fuel storage capacity are acceptable.

3.3 Handling of Heavy Loads and Spent Fuel Assemblies

The rerack modifications proposed by the licensee involve considerations regarding NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980. Specifically, the rerack modifications involve the handling and control of heavy loads, including the movement of spent fuel assemblies, the movement and installation of spent fuel storage racks, the movement of the gates separating the fuel transfer canal and the cask loading pit

from the SFP, the design and use of the hoisting system, safe load paths, the use of procedures, crane operator training, and analyses of postulated load drop accidents and consequences.

3.3.1 Hoisting System

NUREG-0612 provides guidelines and recommendations for licensees to assure safe handling of heavy loads over spent fuel pools and near spent fuel assemblies by prohibiting, to the extent practicable, load travel over spent fuel assemblies, over the core for an operating reactor, and over safety-related equipment. NUREG-0612 defines a heavy load as any load carried in a given area during the operation of the plant that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool for the specific plant in question.

The WCGS Safety Evaluation Report (NUREG-0830, Supplement 3, Appendix G dated May 1984) defines a heavy load weight as 2000 pounds. The maximum weight of the spent fuel rack and its associated rigging is 44,000 pounds. The licensee will use the fuel building 150-ton overhead cask handling crane to remove and install the fuel racks in the pool. As stated in NUREG-0830, Supplement 3 and the licensee's February 24, 1998 submittal, the cask handling crane is designed in accordance with the requirements of CMAA No. 70, "Specifications for Electric Overhead Traveling Cranes," Crane Manufacturers Association of America, Inc., 1975, and ANSI B30.2-1976, "Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)." In addition, a temporary gantry crane and hoist with a capacity of approximately 82,700 pounds will be used to move spent fuel racks to areas within the pool that are not accessible with the overhead crane. The capacity of both the overhead and the temporary gantry cranes far exceed the weight of the loads to be moved.

The licensee has indicated that remotely controlled lifting rigs will be interposed between the crane hook and the racks and used to remove and install fuel racks in the spent fuel pool. The lifting rigs will be designed and tested in accordance with the guidelines in NUREG-0612 and requirements in ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials." The lifting rigs are comprised of independently loaded lift rods such that failure of a single lift rod will not result in uncontrolled lowering of the rack. Both the stress design and the load testing of the lifting rigs satisfy guidelines for upgrading lifting devices in Section 5.1.6(1)(a) in NUREG-0612 and Section 6 in ANSI N14.6-1978. As stated by the licensee, the design of the lifting rig will meet the following:

1. a safety factor of 10 to 1 (twice the normal stress design factor),

2. load tested to three times the maximum weight to be lifted and the load sustained for a minimum of 10 minutes, and

3. following load testing and movement, the integrity of the critical joints and welds of the lifting rig will be examined using liquid penetrant.

In addition to the lifting system, the licensee will use a defense-in-depth approach as outlined in NUREG-0612, Phase 1. Accordingly, the licensee plans to focus on four major causes of load handling accidents (1) operator errors, (2) rigging failures, (3) inadequate inspections, and (4)

inadequate procedures. To preclude potential accidents in these areas, the licensee will perform the following (1) train crane and fuel-bridge operators and other rerack personnel in accordance with ANSI B30.2, (2) use redundantly designed lifting devices that comply with the provisions of ANSI N14.6-1978, and (3) implement inspection and control procedures that govern various rack installation activities.

Further, to enhance heavy load handling operations, the licensee has indicated that additional administrative safety measures will be taken to preclude heavy load drops onto stored spent fuel. These measures include performing inspections and maintenance of the crane prior to the rerack operation, lift testing the old fuel racks prior to moving them, restricting the movement of any racks over or within three feet of active fuel in the SFP (i.e., racks will not be moved over stored fuel and a lateral free clearance of 3 feet will be maintained between the rack being lifted and stored fuel when the rack is lifted above the fuel), separating the upending and laying down of racks from safety-related equipment, and training personnel in the use of upending and lifting equipment. In addition, the licensee will implement controls and procedures to govern crane operation during the transport of the racks.

The staff concludes that the crane, lifting rigs, and the additional measures taken by the licensee to satisfy guidelines in NUREG-0612 will ensure safe lifting of the racks and the gates during the rerack modification. The design and testing of the lifting rigs and other lifting devices coupled with the capacity of the overhead and gantry cranes, and the use of procedures during the rerack operation will enable the licensee to handle heavy loads without undue risks to the safety of the operation. Furthermore, the procedures will help ensure proper establishment and maintenance of the safe load paths and rigging configurations.

3.3.2 Postulated Load Drop Accidents

Generic Letter (GL) 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants,' NUREG-0612," did not require licensees to perform analyses of the consequences of dropping heavy loads described in NUREG-0612, Phase II. However, in GL 85-11, the staff encouraged licensees to implement actions identified in the Phase II analysis regarding the handling of heavy loads considered appropriate to enhance safety. The licensee analyzed the postulated load drop accidents involving the free-fall of a fuel assembly and the gate onto the top of the spent fuel racks, and onto the base plate of the rack modules. A drop of a spent fuel rack from the highest lift point of 40 feet from the bottom of the spent fuel pool was also analyzed. The licensee did not analyze a spent fuel cask drop during the rerack operation even though cask washdown and loading and unloading activities will be staged in the cask washdown pit that is adjacent to the SFP.

As determined by the licensee, the accidental drop of a spent fuel pool rack is highly unlikely. However, a drop of a fuel assembly onto the racks and the pool liner would result in localized deformation of the tip of the racks, no damage to the pool liner, and negligible radiological consequences. A drop of the heaviest rack from the highest lift point to the bottom of the pool would result in piercing of the pool liner and indentation of the concrete below the liner causing minor fluid loss in the pool.

As noted above, a cask drop was not analyzed; however, the licensee plans to use interlocks and procedures to restrict movement and lifting of spent fuel casks over or near fuel in the spent fuel pool, the cask loading pit, the new storage racks or any safety-related equipment. Movement of spent fuel casks over the cask loading pit would only be allowed if the guidelines in NUREG-0612, Phase II are satisfied. The licensee has also indicated that safe plant shutdown would be unaffected due to the physical separation of the fuel building from safe shutdown equipment that is located in the auxiliary, control, and containment buildings. These administrative and procedural measures will enable the licensee to perform its cask operation in accordance with NUREG-0612 guidelines and, therefore, are acceptable to the staff.

Conclusion

Based on the above evaluation, the staff finds that the considerations for the movement of heavy loads in support of the proposed TS changes and expansion of the spent fuel pool storage capacity are acceptable. The licensee's use of the cask handling crane, the temporary gantry crane, and the special lifting devices to move spent fuel racks will be in accordance with the guidelines in NUREG-0612 and ANSI N14.6. Furthermore, the administrative controls and procedures to improve the handling and control of the racks and spent fuel casks will enhance the licensee's capability to reduce the potential for heavy load drop accidents. In addition, although the licensee did not analyze the drop of a spent fuel shipping/storage cask, the licensee's fuel handling accident analyses determined that the consequences of a drop of a rack or spent fuel assembly will not exceed the recommended guidelines in NUREG-0612, Section 5.1.1. Physical and procedural restrictions will preclude movement of spent fuel casks over the spent fuel pool, the cask loading pit and safe shutdown equipment in accordance with existing regulatory guidelines. Therefore, the staff finds that, based on the considerations given to the movement of heavy loads, the proposed modifications to the TS in support of expanding the storage capacity of the spent fuel pool are acceptable.

3.4 Structural Engineering

This evaluation addresses the adequacy of the structural aspects of the licensee's proposal to increase the WCGS spent fuel storage capacity. The primary purpose of this evaluation is to assure the structural integrity and functionality of the racks, the stored fuel assemblies, and the spent fuel pool structure under postulated loads (SRP Section 3.8.4, Appendix D) and fuel handling accidents.

3.4.1 Spent Fuel Storage Racks

The licensee has proposed to increase the spent fuel storage capacity at WCGS to a total of 2642 fuel assemblies. The licensee proposes to store the fuel assemblies in eighteen new fuel storage racks, which are designed as seismic Category I equipment. The new spent fuel storage racks are required to remain functional during and after a safe shutdown earthquake (SSE). Fifteen new storage racks will be placed in the spent fuel pool and three new storage racks will be placed in the cask loading pit. The proposed new spent fuel racks will be designed and manufactured by Holtec International, Inc. (Holtec). The licensee, with its contractor, Holtec, performed structural analyses for the racks in support of the license amendment request.

The licensee used the computer program DYNARACK for dynamic analysis to demonstrate the structural adequacy of the WCGS spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and are not attached to the floor of the storage pool. A nonlinear dynamic model was used to simulate three dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies, including frictional and hydrodynamic effects. The model consisted of inertial mass elements, spring elements, gap elements and friction elements which are defined in the program. 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.

The licensee performed two model analyses: the 3-D single rack (SR) model analysis and the 3-D whole pool multi-rack (MR) model analysis. In the 3-D SR model analysis, the rack was considered fully loaded with a coefficient of friction of 0.8 between the rack and the pool floor. This model analysis investigated the stability of the rack with respect to the overturning. In the 3-D MR analysis, 18 free standing racks were considered fully loaded and half loaded with three different coefficients of friction (μ=0.2, 0.5 and 0.8). This model analysis investigated the fluid-structure interaction effects between racks and pool walls, as well as those among the racks. This model analysis was also used to identify the worst case response for rack movement and for rack member stresses.

The licensee performed the seismic analyses utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration time history) was generated from the design response spectra defined in the WCGS Updated Final Safety Analysis Report. The licensee 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 nineteen 3-D SR and MR analyses were performed. The racks were subjected to the service, upset and faulted loading conditions (Level A, B and D Service Limits). The results of the analyses indicate that the maximum displacements of the racks at the top would be 0.677 inch and 1.274 inches for the SFP and cask loading pit, respectively, indicating that there would be no impacts between spent fuel racks or between the racks and the wall. The results of the analyses also demonstrate that there is adequate safety margin against overturning of the racks. In addition, the licensee compared the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension with the corresponding allowable stresses specified in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. The results show that all induced stresses under the SSE loading condition are less than the corresponding allowable stresses specified in the ASME Boiler and Pressure Vessel Code, indicating that the rack design is adequate.

The licensee also calculated the stresses in the rack weld connections (e.g., baseplate-to-rack, baseplate-to-pedestal and cell-to-cell connections) under the dynamic loading conditions. The licensee demonstrated that the calculated weld stresses are less than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the new spent fuel rack is adequate.

Based on (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when compared to the corresponding allowable stresses provided in the ASME Boiler and Pressure Vessel Code, and (3) the licensee's overall structural integrity conclusions supported by both SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions. Therefore, the spent fuel racks are acceptable.

3.4.2 Spent Fuel Storage Pool

The licensee analyzed the SFP and cask loading pit using the finite element computer program, STARDYNE, to demonstrate the adequacy of the structures under fully loaded fuel racks with all storage locations occupied by fuel assemblies. The fully loaded structures were subjected to the load combinations specified in the WCGS Updated Final Safety Analysis Report.

The licensee assumed a bulk fuel pool temperature of 140°F in the thermal load for the SFP and cask loading pit structural analyses. However, this temperature is less than the temperature of 170°F used in the thermal hydraulics analysis provided in the licensee's application for amendment dated March 20, 1998. The staff identified a concern regarding the higher peak SFP temperature and its potential effect on the structural responses if assumed in the structural analyses. The licensee provided additional information that justified the structural analysis results and the structural analysis assumption of 140°F rather than the higher temperature of 170°F. The justifications provided were (1) a compressive strength of 4000 psi was assumed for the concrete in the analyses while the actual compressive strength of the concrete is approximately 5300 psi, (2) a damping value of 4% was used in the analyses while the SRP allows a higher, less conservative damping value of 7%, and (3) cracked concrete sections were considered, thereby reducing the concrete stiffnesses, in accordance with the ACI Standard 349 methodology used in the analyses. In addition, in the event of a loss of a train of SFP cooling during a full core off load, the licensee demonstrated that the SFP temperature would reach a maximum of 170°F, and would not be greater than 150°F for more than 9 days. In consideration of the conservatisms included in the structural analyses and the licensee's evaluation, the staff finds that the structural analyses is acceptable.

The licensee's application for amendment dated March 20, 1998, included Tables 8.5.1 and 8.5.2. These tables provide the predicted factors of safety, varying from 1.05 to 3.61, for shear force and bending moments of the concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the licensee's structural analyses demonstrate the adequacy and integrity of the structures under full fuel loading, thermal loading and SSE loading conditions. Therefore, the storage fuel pool design is acceptable.

3.4.3 Fuel Handling Accident

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

The analysis results of refueling accident case (1) indicate that the load transmitted to the SFP 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 postulated fuel assembly drop through the rack structure. The licensee's analysis results for refueling accident case (2) indicate that damage will be restricted to a depth of 6.43 inches below the top of the rack, which is above the active fuel region. The staff has reviewed the licensee's analysis results provided in its application for amendment dated March 20, 1998, and concurs with its findings. The results are acceptable based on the licensee's structural integrity conclusions supported by the parametric studies.

3.4.4 Structural Engineering Conclusions

Based on the review and evaluation of the licensee's application for amendment dated March 20, 1998, and supplements, the staff concludes that the licensee's design of the spent fuel rack modules and the spent fuel pool structures are adequate to withstand the effects of the applicable loads, including loads resulting from an SSE. The licensee's structural analysis and design are in compliance with current licensing basis set forth in the WCGS Updated Final Safety Analysis Report and applicable provisions of the SRP and are, therefore, acceptable.

3.5 Materials Engineering

The new high density spent fuel racks to be installed in the SFP are designed and manufactured by Holtec International. The racks are free-standing and self-supporting. The racks are designed to American Society of Mechanical Engineers (ASME) Code stress limits and analyzed in accordance with Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel Code.

3.5.1 Structural Materials

The following structural materials are used in the new spent fuel racks:

(1) Storage cell structures and internally threaded support legs are made from 304L stainless steel, according to ASME Standard SA240.

(2) Externally threaded support spindles are made from precipitation hardened stainless steel (heat treated to 1100F) in accordance with ASME Standard SA564-630.

(3) The welds are of ASME Type 308L material.

These materials have been previously used in many other applications. The materials have been exposed to environments similar to or more severe than those existing in the WCGS SFP without experiencing any observable corrosion damage. They are, therefore, acceptable for their present application.

3.5.2 Neutron Poison Material

Boral is utilized as a neutron absorbing material in the spent fuel racks. Boral is a cermet composite material made of Type 1100 aluminum and boron carbide. The composite panel

consists of boron carbide particles embedded 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. The oxide is stable in environments 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 preventive corrosion properties of Boral, some corrosion is expected. Although this level of corrosion will not usually result in a significant depletion of boron and resultant 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 have not yet formed a protective oxide film. This hydrogen, if not vented, could cause swelling of the sheathings holding Boral panels and resultant deformation of storage cells. In order to prevent this from occurring, the Holtec manufactured racks will have vented Boral sheathings, allowing the generated hydrogen to escape. Production of hydrogen from this process will significantly decrease as the rack aluminum surfaces develop a protective oxide film.

3.5.3 Conclusion

Based on the above evaluation, the staff finds that the materials in the spent fuel racks, manufactured by Holtec International, are compatible with the environment in the WCGS SFP. These new SFP racks will not undergo material degradation which could affect their ability to safely store spent and new fuel. A vented design of the Boral sheathings prevents the corrosion generated hydrogen from building up pressures which could cause distortion of the fuel cells. The staff concludes, therefore, that all the materials used in the new spent fuel racks are acceptable.

3.6 Radiological Protection

3.6.1 Occupational Dose Control

The staff has reviewed the licensee's plans for the modification of the WCGS spent fuel racks with respect to occupational radiation exposure. As stated in Section 2.0, the licensee plans to install a total of 18 new fuel rack modules in the SFP and cask loading pit. A number of facilities have previously performed similar reracking modifications. On the basis of the lessons learned from these modifications, the licensee estimates that the proposed fuel rack installation can be performed with a dose consequence of between 6 and 12 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA (as low as is reasonably achievable) principles. The Radiation Protection Department will prepare Radiation Work Permits (RWPs) for the various jobs associated with the reracking modification. These RWPs will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels (including potential exposure to hot particles), and dosimetry requirements. Each project team member will receive radiation protection training on the reracking modification. Project team members will also be required to attend daily pre-job briefings on the scope of the work to be performed. Personnel will wear protective clothing and be required to wear, at a minimum, personnel monitoring equipment consisting of TLDs and self-reading dosimeters.

If necessary, the licensee will use divers for the removal of the existing SFP rack modules and installation of the replacement high density racks. These divers may also be used to facilitate removal of sparger piping in the SFP. Each diver will be equipped with electronic dosimeters with remote, above surface, readouts which will be continuously monitored by Radiation Protection personnel. Divers will also have access to an underwater survey meter. The diving area will be well illuminated and the divers will be under constant visual surveillance. If needed, TV monitoring will be available to monitor the diver's location and work activities. Divers will also be in continuous communication with Radiation Protection personnel. The licensee will conduct radiation surveys of the diving area prior to each diving operation and following the movement on any irradiated hardware in the SFP. The licensee will use visual barriers (such as air bubbles, ropes, or netting), as practical, in order to minimize diver doses when working near high radiation sources in the SFP. Divers' movements may be restricted by the use of a diver umbilical in order to ensure that divers maintain a safe distance from irradiated sources.

The licensee will use a pressure washer or other acceptable cleaning mechanism to decontaminate the existing SFP rack modules (as well as any interferences or hardware that must be removed) prior to removal from the SFP. Abrasive tools may also be used to supplement the removal of any hot particles. All items removed from the SFP will be closely monitored for hot particles. Once the SFP racks and other hardware are removed from the SFP, they will be rinsed with demineralized water, put in anti-contamination bags, and placed in a special DOT approved shipping container. The removed SFP racks will be shipped offsite for disposal.

The licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase due to the expanded SFP storage capacity. However, the licensee will operate continuous air monitors in areas where there is a potential for significant airborne activity during the fuel reracking modification. In addition, the plant effluent radiation monitoring system will monitor any gaseous releases.

The licensee will monitor and control personnel traffic and equipment movement in the SFP area to minimize contamination and to assure that personnel exposures are maintained ALARA. In order to minimize worker doses, the licensee has developed several remote tools that will be used during the reracking operation. The licensee plans to use an underwater vacuum cleaner system to remove crud and debris from the bottom of the SFP following removal of the old SFP rack modules. This vacuum system will also be used to capture metal filings generated by any cutting performed in the SFP. The licensee will use the existing SFP filtration system during fuel rack installation to maintain water clarity in the SFP.

The storage of additional spent fuel assemblies in the SFP will not increase the dose rates on the refueling floor or in adjacent accessible areas to the SFP. The dose rate at the SFP surface is not expected to increase due to the increased fuel storage and will remain a Zone "C" area (< 10 mrem/hr). Due to storage of spent fuel closer to the SFP wall, which is adjacent to the cask washdown pH, the cask washdown pH is being rezoned from Zone "B" (< 2.5 mrem/hr) to Zone "E" (< 100 mrem/hr). The licensee will implement appropriate access controls for this area to restrict personnel access. The radiation zoning for the pool surface above the cask loading pit will change from Zone "B" to Zone "C" (< 10 mrem/hr) as a result of the storage of spent fuel in this area. In order to ensure that area dose rates in accessible areas adjacent to the SFP and cask loading pit do not exceed the maximum dose rate levels for

which these areas are zoned, the licensee plans include administrative controls which will specify that freshly discharged fuel will not be stored in the cask loading pit or in designated storage cell locations in the SFP.

On the basis of our review of the licensee's proposal, the staff concludes that the WCGS SFP rack modification can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The staff finds the licensee's projected occupational dose of 6 to 12 person-rem for the project to be in the range of doses for similar SFP modifications at other plants, and is reasonable.

3.6.2 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP purification system. These spent resins are changed about once a year at WCGS. The licensee predicts that the resin changeout frequency of the SFP purification system may be increased temporarily during the reracking modification. In order to maintain the SFP water as clean as possible, and thereby minimize the generation of spent resins, the licensee will vacuum the floor of the SFP to remove radioactive crud, sediment, and other debris before the new fuel rack modules are installed. Filters from these underwater vacuums will be a source of solid radwaste. Additional solid radwaste will consist of the old SFP rack modules themselves as well as any interferences or SFP hardware that may have to be removed from the SFP to permit installation of the new spent fuel rack modules. Overall, however, the licensee does not expect that increasing the storage capacity of the SFP will result in a significant change in the generation of solid radwaste at WCGS.

3.6.3 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of a fuel handling accident (FHA) to determine the thyroid and whole-body doses at the exclusion area boundary (EAB), low population zone (LPZ), and control room. The proposed WCGS SFP reracking will not affect any of the assumptions or inputs used in evaluating the dose consequences of the FHA.

The staff reviewed the licensee's analysis and performed confirmatory calculations to check the acceptability of the projected doses. In performing these calculations, the staff used the assumptions of RG 1.25, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." For WCGS, an FHA occurring in the reactor building would result in a higher dose to the control room operator than for an FHA in the fuel building. For an FHA in the reactor building, the staff assumed that the cladding of 317 fuel rods (one full assembly plus 20 percent of the rods of an additional assembly) would be ruptured if a fuel assembly were dropped during handling. The damaged fuel rods are assumed to contain freshly off-loaded fuel with a minimum of 100 hours of decay. The parameters which the staff utilized in its assessment are presented in Table 1.

The staff's calculations confirmed that the thyroid doses at the EAB, LPZ, and in the control room resulting from a postulated FHA meet the acceptance criteria and that the licensee's calculations are acceptable. The results of the staff's calculations are presented in Table 2. For an FHA, the staff calculated a dose of 57.2 rem thyroid at the EAB and 8.2 rem thyroid at

the LPZ. The acceptance criterion at the EAB and LPZ for these accidents is contained in SRP Section 15.7.4 of NUREG-0800 and is 75 rem thyroid dose (25 percent of 10 CFR Part 100 guidelines of 300 rem). The staff calculated a dose to the control room operator of 15.4 rem thyroid. The acceptance criterion for the control room operator is 30 rem thyroid (NUREG-0800, SRP Section 6.4). The staff, therefore, finds the proposed WCGS SFP reracking modification to be acceptable with respect to potential radiological consequences as a result of a postulated fuel handling accident.

4.0 STATE CONSULTATION

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

5.0 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 December 11, 1998 (63 FR 68478). 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.

6.0 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.

Attachments: 1. Table 1
2. Table 2

Principal Contributors: C. Hinson
Y. Kim
L. Kopp
C. Lauron
V. Ordaz
B. Thomas

Date: March 22, 1999

TABLE 1

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT AT WOLF CREEK GENERATING STATION

Parameters	
Power Level, Mwt	3565
Number of Fuel Rods Damaged (1 assembly plus 20%)	317
Total Number of Rods in Core	50,952
Shutdown Time, hours	100
Power Peaking Factor	1.65
Fission-Product Release Fractions (%)*	
Iodine (corrected for extended burnup)	12
Noble Gases	30
Pool Decontamination Factors*	
Iodine	100
Noble Gases	1
Iodine Forms (%)*	
Elemental	75
Organic	25
Filter Efficiencies for Control Room (%)	
Elemental	90
Organic	90
Iodine Protection Factor (IPF)	14
Atmospheric Dispersion Factors, X/Q (sec/m^3)	
Exclusion Area Boundary (0-2 hours)**	1.4×10^{-4}
Low Population Zone (0-8 hours)**	2.0×10^{-5}
Control Room (0-8 hours)**	5.3×10^{-4}
Dose Conversion Factors per ICRP 2	

* Regulatory Guide 1.25
** Staff calculated

TABLE 2

THYROID DOSES FROM FUEL HANDLING ACCIDENT AT WOLF CREEK GENERATING STATION (VALUES CALCULATED BY NRC STAFF)

	DOSE (REM)
	FUEL HANDLING ACCIDENT
EAB*	57.2
LPZ*	8.2
Control Room**	15.4

*Acceptance Criterion = 75 rem thyroid
**Acceptance Criterion = 30 rem thyroid