December 8, 199

50-482

Mr. Neil S. Carns
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M92546)

Dear Mr. Carns:

The Commission has issued the enclosed Amendment No. 92 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 June 14, 1995, as supplemented by letters dated July 13, 1995, and August 22, 1995.

The amendment revises Technical Specification (TS) 3.2.3, "Nuclear Enthalpy Rise Hot Channel Factor," TS 6.9.1.9, "Core Operating Limits Report," and the associated Bases sections. The revisions incorporate changes associated with the planned implementation of advanced nuclear and core thermal-hydraulic design methodologies licensed from Westinghouse Electric Corporation for core reload design, starting with Cycle 9.

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

James C. Stone, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Amendment No. 92 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION

Docket File	OGC, 015B18
PUBLIC	WBateman
PDIV-2 Reading	GHill (2)
EGA1	CGrimes, 011E22
RHuey, RIV	JRoe
JStone	ACRS
EPeyton	CMH2 (SE)
LHurley, RIV	RJones, SRXB

DOCUMENT NAME: WC92546.AMD

OFFICE	PDIV-2/LA	PDIV-2/PM	OGC
NAME	EPeyton	JStone:pk	
DATE	11/15/95	11/16/95	12/5/95

OFFICIAL RECORD COPY

140042

DFO1

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

December 8, 1995

Mr. Neil S. Carns
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M92546)

Dear Mr. Carns:

The Commission has issued the enclosed Amendment No. 92 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 June 14, 1995, as supplemented by letters dated July 13, 1995, and August 22, 1995.

The amendment revises Technical Specification (TS) 3.2.3, "Nuclear Enthalpy Rise Hot Channel Factor," TS 6.9.1.9, "Core Operating Limits Report," and the associated Bases sections. The revisions incorporate changes associated with the planned implementation of advanced nuclear and core thermal-hydraulic design methodologies licensed from Westinghouse Electric Corporation for core reload design, starting with Cycle 9.

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,

James C. Stone

James C. Stone, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Amendment No. 92 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

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

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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

Public Health Physicist
Bureau of Air & Radiation
Division of Environment
Kansas Department of Health
and Environment
Forbes Field Building 283
Topeka, Kansas 66620

Vice President Plant Operations
Wolf Creek Nuclear Operating Corporation
P. O. Box 411
Burlington, Kansas 66839

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

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 65077-1302

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

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. 92
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 June 14, 1995, as supplemented by letters dated July 13, 1995, and August 22, 1995, 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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 92, 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 and shall be implemented prior to restart from the eighth refueling outage, which is scheduled to begin in March 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

James C. Stone

James C. Stone, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 92

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 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	---
3/4 2-10	---
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
6-21	6-21
6-21a	6-21a
6-21b	6-21b

THIS PAGE HAS BEEN DELETED

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \le \frac{[F_Q^{RTP}]}{P} [K(Z)] \text{ for } P > 0.5 \text{, and}$$

$$F_Q(Z) \le \frac{[F_Q^{RTP}]}{0.5} [K(Z)] \text{ for } P \le 0.5.$$

Where:

F_Q^{RTP} = the F_Q (Z) Limit at RATED THERMAL POWER (RTP), as specified in the CORE OPERATING LIMITS REPORT (COLR),

$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the normalized $F_Q(Z)$ limit as a function of core height, as specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 8 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER;

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify that the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(Z) \le \frac{[F_Q^{RTP}][K(Z)]}{[P][W(Z)]} \quad \text{for } P > 0.5$$

$$F_Q^M(Z) \le \frac{[F_Q^{RTP}][K(Z)]}{[0.5][W(Z)]} \quad \text{for } P \le 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty and $W(Z)$ is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is provided in the COLR.

d. Measuring $F_Q^M(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding, by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right]$$

has increased since the previous determination of $F_Q^M(Z)$, either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased over that specified in 4.2.2.2.c by an appropriate factor specified in the COLR, or

*During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved after which a power distribution map may be obtained.

SURVEILLANCE REQUIREMENTS (Continued)

2. $F_Q^M(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right] \text{ is not increasing.}$$

f. With the relationships specified in 4.2.2.2.c above not being satisifed:

1. Calculate the percent $F_Q^M(Z)$ exceeds its limit by the following expression:

$$\left\{ \left(\frac{\text{maximum}}{\text{over } z} \left[\frac{F_Q^M(Z) \text{X } W(Z)}{\frac{F_Q^{RTP}}{P} \text{ X } K(Z)} \right] \right) -1 \right\} \text{ X } 100 \text{ for } P \geq 0.5$$

$$\left\{ \left(\frac{\text{maximum}}{\text{over } z} \left[\frac{F_Q^M(Z) \text{X } W(Z)}{\frac{F_Q^{RTP}}{0.5} \text{ X } K(Z)} \right] \right) -1 \right\} \text{ X } 100 \text{ for } P < 0.5$$

2. Either one of the following actions shall be taken:

a. Within 2 hours, control the AFD to within new AFD limits which are determined by tightening both the negative and positive AFD limits of Specification 3.2.1 by 1% AFD for each percent $F_Q^M(Z)$ exceeds its limit and declare the AFD monitor alarm inoperable until the AFD alarm setpoints are changed to the modified limits, or

b. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated above.

g. The limits in Specification 4.2.2.2.c, 4.2.2.2.e and 4.2.2.2.f are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive,

2. Upper core region from 85 to 100%, inclusive,

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq F^{RTP}_{\Delta H} [1.0 + PF_{\Delta H} (1.0 - P)]$$

where,

$F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the Core Operating Limits Report (COLR).

$PF_{\Delta H}$ = The power factor multiplier for $F^N_{\Delta H}$ specified in the COLR,

$P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used since an uncertainty of 4% for incore measurement of $F^N_{\Delta H}$ has been included in the above limit.

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 4 hours, either

   1. Restore $F^N_{\Delta H}$ to within the above limit, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 72 hours of initially being outside the above limit, verify through incore flux mapping that $F^N_{\Delta H}$ has been restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by Actions a. or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core flux mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days, and

c. The provisions of Specification 4.0.4 are not applicable.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to the DNBR design limit specified in the CORE OPERATING LIMITS REPORT (COLR) during normal operation and in short-term transients, and (b) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux,

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ and $F^N_{\Delta H}$ limits are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD limits have been adjusted for measurement uncertainty.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the AFD limits and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position,

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The limits on the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$ are specified in the COLR.

$F_Q(Z)$ and $F^N_{\Delta H}$ are measured periodically to provide assurance that they remain within their limits. A peaking margin calculation is performed, when necessary, to provide the basis for reducing THERMAL POWER or for reducing the width of the AFD limits. The hot channel factor $F^M_Q(Z)$ is measured periodically and increased by a cycle and height dependent factor, W(Z), to provide assurance that the limit of $F_Q(Z)$ is met. W(Z) accounts for the effects of normal operation transients and is determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(Z) functions are specified in the Core Operating Limits Report.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective ACTION is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

CORE OPERATING LIMITS REPORT (COLR)

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle, for the following:

1. Specification 3.1.1.3: Moderator Temperature Coefficient (MTC) EOL limits

2. Specification 3.1.3.5: Shutdown Rod Insertion Limit

3. Specification 3.1.3.6: Control Rod Insertion Limits

4. Specification 3.2.1: Axial Flux Difference (AFD)

5. Specification 3.2.2: Heat Flux Hot Channel Factor - $F_Q(Z)$

6. Specification 3.2.3: Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$

7. Specification 3.9.1.b: Refueling Boron Concentration

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

a. NRC Safety Evaluation Report dated October 29, 1992, for the "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station" (ET-90-0140, ET 92-0103)

   (Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$).

b. NRC Safety Evaluation Report dated January 17, 1989, for the "Acceptance for Referencing of Licensing Topical Report WCAP-11397, Revised Thermal Design Procedure."

   (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$).

c. NRC Safety Evaluation Report dated September 30, 1993, for the "Transient Analysis Methodology for the Wolf Creek Generating Station" (ET-91-0026, ET 92-0142, WM 93-0010, WM 93-0028)

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient [MTC])

d. NRC Safety Evaluation Report dated November 26, 1993, "Acceptance for Referencing of Revised Version of Licensing Topical Report WCAP-10216-P-A, Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification" (TAC No. M88206).

CORE OPERATING LIMITS REPORT (COLR) (Continued)

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor - $F_Q(Z)$: Specification 3.1.1.3 - Moderator Temperature Coefficient (MTC): Specification 3.1.3.5 - Shutdown Rod Insertion Limit: Specification 3.1.3.6 - Control Rod Insertion Limits: Specification 3.2.1 - Axial Flux Difference: Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$: Specification 3.9.1.b - Refueling Boron Concentration).

e. NRC Safety Evaluation Report dated March 10, 1993, for the "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station" (ET 92-0032, ET 93-0017)

(Methodology for Specification 3.1.3.6 - Control Rod Insertion Limits; Specification 3.2.1 - Axial Flux Difference)

f. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7" (NA 92-0073, NA 93-0013, NA 93-0054)

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$ [Use of WRB-2 Correlation with VIPRE-01 Code])

g. NRC Safety Evaluation Report dated November 13, 1986, for "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code" (WCAP-10266-P-A, Rev. 2)

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor - $F_Q(Z)$)

h. NRC Safety Evaluation Report dated May 17, 1988, "Acceptance for Referencing of Westinghouse Topical Report WCAP-11596 - Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor - $F_Q(Z)$: Specification 3.1.1.3 - Moderator Temperature Coefficient (MTC): Specification 3.1.3.5 - Shutdown Rod Insertion Limit: Specification 3.1.3.6 - Control Rod Insertion Limits: Specification 3.2.1 - Axial Flux Difference: Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$: Specification 3.9.1.b - Refueling Boron Concentration).

i. NRC Safety Evaluation Report dated June 23, 1986, "Acceptance for Referencing of Topical Report WCAP 10965-P and WCAP 10966-NP-ANC: A Westinghouse Advanced Nodal Computer Code."

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor - $F_Q(Z)$: Specification 3.1.1.3 - Moderator Temperature Coefficient (MTC): Specification 3.1.3.5 - Shutdown Rod Insertion Limit: Specification 3.1.3.6 - Control Rod Insertion Limits: Specification 3.2.1 - Axial Flux Difference: Specification 3.2.3 -

CORE OPERATING LIMITS REPORT (COLR) (Continued)

Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$: Specification 3.9.1.b - Refueling Boron Concentration).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-hydraulic limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

RECORD RETENTION (Continued)

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those Unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the Unit Staff;

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of Quality Assurance activities required by the QA Manual;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the PSRC and the NSRC;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality; and

n. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 92 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 June 14, 1995, as supplemented by letters dated July 13, 1995, and August 22, 1995, Wolf Creek Nuclear Operating Corporation (WCNOC or the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station (WCGS). The proposed changes would revise TS 3.2.2, "Heat Flux Hot Channel Factor," TS 3.2.3, "Nuclear Enthalpy Rise Hot Channel Factor," TS 6.9.1.9, "Core Operating Limits Report," and associated Bases sections. WCNOC submitted a safety evaluation report to support the TS change request. This report describes the core nuclear design methods and the core thermal-hydraulic methods used for reload design at the WCGS. The report also documents the capability of WCNOC to perform in-house core reload design analyses for the WCGS using standard Westinghouse Electric Corporation (W) methodologies approved by the NRC.

The NRC has previously approved the use by WCNOC of nuclear and thermal-hydraulic design methods licensed from the Babcock & Wilcox Fuel Company (BWFC) and additional technology licensed from W. WCNOC intends to replace the methods previously used for the core design and reactor physics calculations for Cycle 7 and 8 at WCGS with the currently approved W methodology and computer programs. The following section presents the NRC staff evaluation of the proposed TS changes and the implementation of the new W methodologies.

The August 22, 1995, supplemental letter forwarded the nonproprietary version of the safety evaluation and analysis provided in the June 14, 1995 submittal and did not change the staff's original no significant hazards determination published in the Federal Register on August 2, 1995 (60 FR 39456).

2.0 EVALUATION

The licensee's submittal describes the enhanced W computer programs and models used by WCNOC to analyze reload cores and compares the model predicted results with measurements obtained from benchmarking data covering WCGS operating Cycles 6, 7, and 8. The plant analyses were performed over a range of conditions from hot zero power (HZP) to hot full power (HFP). The agreement between the measured and calculated values presented is used to validate the application of the computer programs and the ability of WCNOC to perform analysis of the WCGS.

WCNOC intends to use these methods for steady-state core reload design applications, including nuclear, thermal-hydraulic, and non-LOCA (loss-of-coolant accident) safety analyses, beginning with Cycle 9 at WCGS.

Nuclear Design

The NRC-approved W reactor physics code system ALPHA/PHOENIX/ANC (APA) has been installed at WCGS. The primary physics codes included in this system are PHOENIX-P, ANC, and APOLLO. PHOENIX-P is a two-dimensional multigroup transport code for generating lattice physics constants. ANC is a nodal code used mainly for three-dimensional core design calculations. APOLLO is a two-group, one-dimensional neutron diffusion code.

WCNOC has benchmarked these codes against plant measurements from WCNGS Cycles 6, 7, and 8. These benchmarks include control rod worths, critical boron concentrations, and moderator temperature coefficients (MTCs) from startup physics tests, as well as boron letdown curves from core follow operation and power distributions from monthly incore flux maps. These comparisons indicate that the difference between the measured and predicted values meets the physics test acceptance criteria for all cases. The staff, therefore, concludes that WCNOC has appropriately demonstrated their ability to properly use the W physics methods. WCNOC has stated that measurements of physics parameters during initial cycle startup and during cycle operation will continue to be compared to predicted values to verify the validity of the uncertainties assigned to reload safety parameters and the accuracy of the physics predictions.

Thermal-Hydraulic Design

The core thermal-hydraulic analysis for each reload design is performed with the VIPRE-01 code. WCNOC has previously received NRC approval of modeling methodologies, heat transfer correlation selection, and departure from nucleate boiling ratio (DNBR) limit for VIPRE-01 models of the WCGS core. The WCGS Cycle 9 VIPRE-01 thermal-hydraulic model represents a full core load of the W Vantage 5 Hybrid (V5H) with intermediate flow mixers (IFM) fuel design. Several changes were made to the Cycle 8 base thermal-hydraulic model to facilitate the transition to W nuclear and thermal-hydraulic methodologies. WCNOC has provided the appropriate bases as well as the implementation details to justify and demonstrate the adequacy of these changes for calculating thermal-hydraulic conditions within the hot assembly.

Cycles 7 and 8 of the WCGS core thermal-hydraulic analysis employed the BWFC statistical core design (SCD) methodology. Beginning with Cycle 9, the analysis will be based on the NRC-approved W revised thermal design procedure (RTDP). The staff has reviewed the WCGS uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters, and in the thermal-hydraulic and transient codes that were statistically combined with the DNBR correlation uncertainties to obtain an overall DNBR uncertainty factor and finds them acceptable. The uncertainty factor was used to obtain the design limit DNBR of 1.23 for Cycle 9 operation. This design limit satisfies the thermal-hydraulic design basis, which is to protect against DNB such that there is at least a 95 percent probability at the 95 percent confidence level that a DNB will not occur during normal operation or anticipated operational occurrences. In order to produce sufficient margin for core design flexibility, and to offset penalties such as those due to rod bow, lower plenum flow anomaly, and transition cores, the design limit DNBR is increased to a value designated as the safety analysis limit DNBR. The staff has reviewed the safety analysis limit DNBR and concludes that it was set high enough to cover all known DNBR penalties plus an additional 9 percent retained margin for flexibility.

Technical Specification Changes

The licensee proposes revising TS 3.2.2, "Heat Flux Hot Channel Factor," and TS 3.2.3, "Nuclear Enthalpy Rise Hot Channel Factor." The transition to W computer codes and methodology and the use of relaxed axial offset control (RAOC) for power distribution monitoring necessitate the implementation of compatible W TS for monitoring the heat flux hot channel factor (F_Q) and the nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$). The revisions are analogous to current W TS for plants using RAOC and incorporate the calculation of NRC-approved parameters located in the Core Operating Limits Report (COLR).

The licensee has also proposed revisions to TS 6.9.1.9 to change the nomenclature for the Heat Flux Hot Channel Factor and Nuclear Enthalpy Rise Hot Channel Factor and to reference the analytical methods to be used to calculate the core operating limits. These additional methods are:

1. NRC Safety Evaluation Report dated January 17, 1989, for the "Acceptance for Referencing of Licensing Topical Report WCAP-11397, Revised Thermal Design Procedure."

2. NRC Safety Evaluation Report dated November 26, 1993, "Acceptance for Referencing of Revised Version of Licensing Topical Report WCAP-10216-P-A, Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification."

3. NRC Safety Evaluation Report dated May 17, 1988, "Acceptance for Referencing of Westinghouse Topical Report WCAP-11596 - Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

4. NRC Safety Evaluation Report dated June 23, 1986, "Acceptance for Referencing of Topical Report WCAP 10965-P and WCAP 10966-NP-ANC: A Westinghouse Advanced Nodal Computer Code."

The licensee has proposed deleting the reference to: NRC Safety Evaluation Report dated March 26, 1993, for the "Qualification of the Steady State Core Physics Methodology for the Wolf Creek Generating Station" (ET 92-0011, WM 93-0038). This methodology will no longer be used by WCNOC.

3.0 SUMMARY AND CONCLUSIONS

WCNOC has been performing nuclear and core thermal-hydraulic analyses for the WCGS using both BWFC and W design methodologies. Beginning with the Cycle 9 reload design, WCNOC intends to use only W advanced nuclear and core thermal-hydraulic design technology. The methods employed and described in the topical report are standard W methods and reflect current practices. WCNOC has used the NRC-approved W methodology to model operating Cycles 6, 7, and 8, and has performed detailed comparisons of the results to measured operating data. In general, the WCNOC predictions agreed well with measurements. All startup test predictions fell within the required review and acceptance criteria. In addition, comparisons between power operation measurements and WCNOC predictions for boron rundown, peaking factors, and power distributions showed good agreement. This effort demonstrated the capability of WCNOC to use the W computer program package for application to the WCGS, using the W RAOC power distribution control limit calculational procedure. Measurements of physics parameters during initial cycle startup and during cycle operation will continue to be compared to predicted values to verify the validity of the uncertainties assigned to reload safety parameters and the accuracy of the physics predictions.

W has provided training in the proper use of these codes to the WCNOC core design staff. In addition, W has reviewed the core models of WCGS Cycles 1-8 generated by the WCNOC core design staff. WCNOC has stated that they will use these methods without modifications and in accordance with W training and the approved application of the methods. The NRC concludes that an acceptable training program was implemented to ensure that WCNOC personnel have a good working knowledge of the codes and methods, will be able to set up the input, understand and interpret the output results, understand the applications and limitations, and perform analyses in compliance with the application procedure.

Based on the analyses and results presented in the topical report, the staff concludes that the W methodology, as validated by WCNOC, can be applied by WCNOC to steady-state nuclear design and thermal-hydraulic calculations for the WCGS reload design applications discussed in the above technical evaluation. The accuracy of this methodology has been demonstrated to be sufficient for use in design applications, including PWR reload physics and thermal-hydraulic analysis, core limit generation and protection, and reload safety analysis integration. Therefore, the new methodology will ensure that the values for cycle specific parameters are determined such that safety and design limits are not exceeded.

The staff finds the proposed TS changes incorporating NRC-approved methologies and use of the enhanced W computer programs and models by WCNOC acceptable.

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

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 39456). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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.

Principal Contributor: L. Kopp

Date: December 8, 1995