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

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