WOLF CREEK
NUCLEAR OPERATING CORPORATION

Karl A. (Tony) Harris
Manager Regulatory Affairs

MAR 11 2003

RA 03-0025

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket No. 50-482: Wolf Creek Generating Station Annual 50.59 Evaluation Report

Gentlemen:

This letter transmits the Annual 50.59 Evaluation Report for Wolf Creek Generating Station (WCGS), which is being submitted pursuant to 10 CFR 50.59(d)(2). Attachment I provides a summary of the evaluation results. Attachment II provides the WCGS Annual 50.59 Evaluation Report.

This report covers the period from January 1, 2002, to December 31, 2002, and contains a summary of 50.59 evaluations performed during this period that were approved by the WCGS onsite review committee. In accordance with the reporting requirement contained in 10 CFR 50.59(d)(2), Wolf Creek Nuclear Operating Corporation will submit future 50.59 evaluation reports at intervals not to exceed 24 months.

There are no commitments contained in this correspondence.

If you have any questions concerning this matter, please contact me at (620) 364-4038, or Ms. Jennifer Yunk at (620) 364-4272.

Very truly yours,

Karl A (Tony) Harris

Karl A. (Tony) Harris

KAH/rlg

Attachments

cc: J. N. Donohew (NRC), w/a
D. N. Graves (NRC), w/a
E. W. Merschoff (NRC), w/a
Senior Resident Inspector (NRC), w/a

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WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station

Docket No.: 50-482
Facility Operating License No.: NPF-42

ANNUAL 50.59 EVALUATION REPORT

Report No.: 18

Reporting Period: January 1, 2002 through December 31, 2002

SUMMARY

This report provides a brief description of changes, tests, and experiments performed at Wolf Creek Generation Station and evaluated pursuant to 10 CFR 50.59(c)(1). This report includes summaries of the associated 50.59 evaluations that were reviewed and found to be acceptable by the Wolf Creek Generating Station (WCGS) onsite review committee for the period beginning January 1, 2002 and ending December 31, 2002. This report is submitted in accordance with the requirements of 10 CFR 50.59(d)(2).

On the basis of these evaluations of changes:

- There is less than a minimal increase in the frequency of occurrence of an accident previously evaluated in the Updated Final Safety Analysis Report (USAR).
- There is less than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the USAR.
- There is less than a minimal increase in the consequences of an accident previously evaluated in the USAR.
- There is less than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the USAR.
- There is no possibility for an accident of a different type than any previously evaluated in the USAR being created.
- There is no possibility for a malfunction of a SSC important to safety with a different result than any previously evaluated in the USAR being created.
- There is no result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered.
- There is no result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses.

Therefore, all items contained within this report have been determined not to require a license amendment.

Evaluation Number: 59 2001-0024 Revision: 0
Title: SGTR Transient and Radiological Response Based on Revised Operator Action Times

Activity Description:
The activity is a reanalysis, both transient response and radiological response, of the Updated Final Analysis Report (USAR) Chapter 15 steam generator tube rupture (SGTR) accident scenario, as documented in calculations AN-99-025, "Steam Generator Tube Rupture Overfill Analysis with Revised Operator Action Times," Revision 1 and AN-01-006, "Radiological Consequences of a Steam Generator Tube Rupture," Revision 0, respectively.

Calculation AN-99-025 Revision 1 presents the revised SGTR scenario with forced overfill and water relief through an assumed stuck-open safety valve, as part of the disposition of Performance Improvement Request (PIR) 2000-1105, pertaining to revised operator action times.

The revised operator action times used in the analysis were chosen to primarily encompass the response times determined for actual SGTR runs performed on the simulator by several Wolf Creek Generating Station (WCGS) operating crews. Operator action times revised in the analysis included changing the time to terminate the Auxiliary Feedwater System (AFW) from the Turbine-Driven Auxiliary Feedwater (TDAFW) pump to the ruptured steam generator from six to eight minutes subsequent to the safety injection (SI) signal and revising the time to initiate the first Reactor Coolant System (RCS) depressurization from five to eight minutes subsequent to termination of the RCS cooldown. Calculation AN-99-025 Revision 1 supersedes the current licensing basis SGTR overfill analysis. Note: Historically, the analysis of the above described SGTR case supports changes to SGTR Emergency Operating Procedure EMG E-3, "Steam Generator Tube Rupture."

The associated calculation AN-01-006 Revision 0, as part of the disposition of PIR 2000-1105, presented a determination of the radiological consequences of the revised SGTR transient analysis, presented in calculation AN-99-025 Revision 1, to ascertain that acceptable acceptance criteria are satisfied, using a source term and the transient analysis output. The radiological consequences calculated included a determination of radiological isotopes released to the atmosphere and the offsite dose consequences to a time of postulated residual heat removal cut-in conditions. Both calculation packages supersede the current licensing basis SGTR transient and radiological consequence analyses and provide the basis and information of this USAR change request.

50.59 Evaluation:
The overall effect of the input changes on the transient analysis results is that they are comparable with those of the current licensing basis. This is expected as the input changes; i.e., the operator action times were minor. The overall effect of the input changes on any limit or acceptance criteria is obtained from the SGTR radiological consequence analysis. Consistent with current licensing basis methodology two cases are analyzed. The Case 1 (pre-accident iodine spike) Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) results are 57.12 and 7.92 rem to the thyroid, respectively, compared to the current licensing basis of 50.02 and 6.96 rem to the thyroid. The Case 2 (concurrent iodine spike) EAB and LPZ results are 22.50 and 3.13 rem to the thyroid, respectively, compared to the current licensing basis of 20.02 and 2.62 rem to the thyroid.

The calculated radiological consequence increase due to the operator action time changes is less than 10% of the margin to the 10 CFR 100 guidelines, which is not exceeded in both cases, and the Standard Review Plan guideline of 30 rem thyroid for Case 2 is not exceeded. Therefore, the acceptance criteria are satisfied and the minimal increase in consequences threshold is not met.

Evaluation Number: 59 2002-0001 Revision: 0
Title: Updating a USAR Described Evaluation Methodology Regarding Post-LOCA Subcriticality

Activity Description:
This activity updates a USAR described evaluation methodology that is used in establishing post-loss of coolant accident (LOCA) subcriticality. Details on the method for assuring post-LOCA subcriticaliy are not found in the current USAR. The connection to the USAR is through reference to licensed Westinghouse Emergency Core Cooling System (ECCS) Methodology reported in WCAP-8339 and WCAP-8471-P-A. These USAR methodology references imply that control rods are not credited when assuring post-LOCA subcriticality. To address the potential for recriticality due to sump dilution when realigning to hot leg recirculation, the licensing basis for WCGS will now credit the negative reactivity boron worth of inserted control rods and xenon at the time of hot leg switchover. This action is in response to issues identified in Westinghouse NSAL-94-016 Revision 2. WCAP-15704 will be used to justify control rod insertion for cold leg breaks.

50.59 Evaluation:
The regulatory review of this activity conservatively concluded that the activity affects post-LOCA subcriticality methodology described in USAR references. The 50.59 Evaluation reviewed the "The Method of Assuring Post-LOCA Subcriticality" and the "Method for Demonstrating Control Rod Insertion in Post-LOCA Evaluations" and concluded that these methodologies are either "previously approved by the NRC" or "not described in the USAR or its references." Therefore, in accordance with section 4.3.8 of NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," the proposed activity does not result in a departure from a method of evaluation described in the USAR and this activity may be implemented without prior Nuclear Regulatory Commission approval.