WOLF CREEK
NUCLEAR OPERATING CORPORATION

Gautam Sen
Manager Regulatory Affairs

March 10, 2011

RA 11-0031

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

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

Gentlemen;

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

This report covers the period from January 1, 2009, to December 31, 2010, and contains a summary of 50.59 evaluations performed during this period that were approved by the WCGS onsite review committee.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4175, or Diane Hooper at (620) 364-4041.

Sincerely,

Diane M. Hooper for

Gautam Sen

GS/rlt

Attachment

cc: E. E. Collins (NRC), w/a
G. B. Miller (NRC), w/a
B. K. Singal (NRC), w/a
Senior Resident Inspector (NRC), w/a

IE47
NRR

WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station

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

BIENNIAL 50.59 EVALUATION REPORT

Report No.: 22

Reporting Period: January 1, 2009 through December 31, 2010

SUMMARY

This report provides a brief description of proposed changes, tests, and experiments that were evaluated for implementation at Wolf Creek Generating Station (WCGS) pursuant to 10 CFR 50.59(c)(1). This report includes summaries of the associated 10 CFR 50.59 evaluations for the period beginning January 1, 2009 and ending December 31, 2010. 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 2009-0002 Revision: 0
Title: Installation of New Feedwater Check Valves

Activity Description:
Four new main feedwater check valves will be installed into the Main Feedwater system. The new check valves are a nozzle style to be located in Area 5 of the Auxiliary Building. The existing check valves are of a swing style located in the Containment Building.

The existing feedwater check valves will remain installed with the valve discs and associated internals removed from these valves. These existing feedwater check valves will no longer serve the function of a check valve. The design function to maintain a pressure boundary, maintain structural integrity, provide a flow path for addition of Auxiliary Feed Water (AFW) for Reactor Coolant System (RCS) cool down under emergency shutdown conditions, and the portion of the system required to function following a Design Basis Accident (DBA) to achieve and maintain safe shutdown remain valid for these valves.

50.59 Evaluation:
With the new check valves located upstream of the auxiliary feedwater connection, the piping volume between the intact steam generators and the check valves remains filled with normal feedwater following a pipe rupture while the feedline piping upstream of the check valves empties out the break. Once auxiliary feedwater is initiated, the feedwater in the piping between the steam generators and the check valves is first purged into the steam generators and then the lower temperature auxiliary feedwater is delivered to the intact steam generators. Feedwater delivery to the intact steam generators is delayed by the amount of time it takes to purge the hotter normal feedwater that remains between the steam generators and the check valves.

The new location of the check valves requires a larger volume of pipe to be purged, increasing the purge time in the affected accident analyses, prior to delivering cooler AFW. However, the increased purge time would be roughly offset by the absence of the time required to refill the piping volume between the check valves and the Main Feedwater (MFW) Isolation Valves (MFIVs) before auxiliary feedwater will be delivered to the intact steam generators.

The reanalyses of the affected Loss of Normal Feedwater (LONF), Loss of Non-emergency AC Power (LONEAC) and Feedwater Line Break (FLB) transients demonstrate that the accident criteria are maintained with almost identical margin, as compared to the current analyses of record. In addition, the new water hammer analysis provides a new forcing function that is incorporated into the pipe stress analysis to verify that all stresses and support loadings are acceptable.

Evaluation Number: 59 2009-0003 Revision: 0
Title: System Descriptions M-10BN and M-10EN Changes

Activity Description:
Due to a non-conservative temperature span for the Refueling Water Storage Tank (RWST), the System Descriptions M-10BN, "Borated Refueling Water Storage System," and M-10EN, "Containment Spray System (CSS)," are being revised.

The change to M-10BN is a revision to the RWST Volumes and Setpoint Information. The minimum and maximum values for injection volumes, Emergency Core Cooling System (ECCS) Pump Transfer time allowance and CSS Pump transfer time allowance have changed.

The change to M-10EN is to reflect the decrease in the time available for operator action to transfer the suction of the CSS pumps from the RWST to the Containment Recirculation Sump.

These changes result in decreasing the minimum time available for operator action for switchover of the ECCS pumps and CSS pumps from injection mode to recirculation mode following a design basis limiting accident. The available operator action time for ECCS pump switchover decreased from 8.3 minutes 8.15 minutes, and for the CSS pumps switchover decreased from 2.59 minutes to 2.18 minutes.

50.59 Evaluation:
The decrease in the time available for operator action for switchover of ECCS pumps from injection mode to recirculation mode is minimal and has no adverse impact on the ECCS function, as operating crews have consistently performed the switchover within the new allowed times. Four Operating crews were timed on the Wolf Creek Generating Station simulator and each crew conducted the switchover within the new allowed times. The design function of the ECCS and CSS will continue to be met as required.

Evaluation Number: 59 2010-0001 **Revision: 0**
Title: Redesign RCS Letdown Orifices and Throttle Valves

Activity Description:
The Reactor Coolant System (RCS) letdown throttle valves are being removed. In addition, the orifice bore diameter for the letdown flow orifices will be reduced by installing orifices that will take the entire pressure drop previously shared by the combination of the orifices and the throttle valves. The change is required because the plant has experienced lower than designed pressure drop across the three letdown orifices and it has caused accelerated wear on the three corresponding throttle valves. Since the smaller orifices will make it more difficult for normal RCS letdown flow to reach the maximum allowed flow of 120 gpm during low power modes of operation, a bypass line with two manual isolation valves is being added around the letdown orifice to increase the flow capability during cool down and heat up operations with low RCS pressure. The RCS letdown throttle valves being removed are for pressure reduction only, they do not perform any isolation function.

Because the installation of a two-inch bypass line around the 45 gpm letdown orifice creates a flow path that could potentially allow a higher letdown flow rate than that assumed in the USAR, the bypass line will be procedurally/administratively isolated, using redundant locked closed valves, during normal operations.

50.59 Evaluation:
The normal letdown portion of the Chemical and Volume Control System (CVCS) provides a path through which a portion of reactor coolant is letdown and directed to the CVCS demineralizers for purification, clean up or discharge. The normal flow rate is 75 gpm with up to 120 gpm assumed in the USAR. The USAR assumes letdown flow during normal operation is limited to 120 gpm to maintain dose rates associated with a potential letdown line break outside containment to less than a small fraction of the guideline doses in 10CFR Part 100.

The flowrate will be unchanged by this modification. The change to the orifice size and deletion of the throttle valves will have no impact upon the design function. Administrative controls will assure that the bypass line will be isolated through redundant locked closed valves when required. This will assure that the letdown flow will not exceed the flow assumed in the USAR. This modification will be done in accordance with applicable NRC requirements, design, material, and construction codes and standards.

Evaluation Number: 59 2010-0003 **Revision: 0**
Title: Installation of Turbine Driven Auxiliary Feedwater Pump Standby Tanks

Activity Description:
Three Safety Related standby condensate water accumulator tanks are being installed above the vestibule in the Turbine Driven Auxiliary Feedwater (TDAFW) Pump Room. The three tanks are vented to atmosphere, self-leveling and are gravity fed by the existing piping from the Condensate Storage Tank (CST). The function of the standby tanks is to provide a temporary supply of demineralized water to the TDAFW Pump during swap over to the Essential Service Water (ESW) system if the CST has been lost due to a Design Basis event (such as a tornado) coincident with a loss of off-site power. The tanks are designed to supply demineralized water only for the duration of time until the safety related ESW source is available following the loss of the CST. The change is being made to ensure that the TDAFW pump will always have an adequate suction source during the most severe conditions.

50.59 Evaluation:
The Auxiliary Feedwater system is relied upon for normal plant shutdown and to respond to several Condition II Faults as well as Condition IV Limiting Faults previously evaluated in the USAR. The loss of non-emergency AC power to the station auxiliaries, loss of normal feedwater flow, loss-of-coolant accident, feedwater line break, steam line break, and steam generator tube rupture accidents were reviewed. The addition of the standby accumulator tanks, as well as the vent and the check valves to the Auxiliary Feedwater system will be supportive of the system's function to supply an emergency supply of cooling water to the steam generators. Redundancy is provided in the design to assure that a single failure will not prevent the plant systems from performing their intended safety function. There is no adverse impact on the room flooding analysis because the volume of water stored in the new components is small in comparison to the previously assumed flooding source and the existing flooding analysis is still bounding. This modification will be designed and installed in accordance with applicable NRC requirements, design, material, and construction codes and standards.

Evaluation Number: 59 2010-0004 **Revision: 0**
Title: Cycle 18 Reload Design Changes

Activity Description:
The Cycle 18 Core Operating Limits Report (COLR) is being revised from Rev. 0 to Rev. 1. The revision to the COLR updates the descriptions of methodology for monitoring compliance with Technical Specification (TS) limits on F_Q and $F^N\Delta H$. This update to the COLR was needed for implementation of the power distribution monitoring system (PDMS) using the Westinghouse code BEACON. The PDMS system monitors compliance with $F^N\Delta H$, F_Q steady state and F_Q transient limits. The base methodology for F_Q surveillance methodology is described in WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification." The associated TS changes were approved and issued as documented in License Amendment No. 188.

This evaluation addresses the following issues.

1) The change in evaluation methodology resulting from use of WCAP-12472-P-A, Addendum 1-A. This is needed as WCAP-12472-P-A, which is implemented in the current BEACON process did not address Addendum 1-A in License Amendment No. 188.

2) To document a variable uncertainty to be applied to transient F_Q margin calculations. BEACON currently uses a fixed uncertainty for transient F_Q margin calculations.

3) Translation of measured power distribution data to Hot Full Power, All Rods Out, Equilibrium Xenon conditions for use in transient F_Q determination.

50.59 Evaluation:
Addendum 1-A to WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System" incorporates fixed incore detectors and an enhanced core neutronics model into the original BEACON system. Only the neutronics model change is relevant to WCGS. This 10 CFR 50.59 evaluation documents that the use of the enhanced neutronics model, as described in Addendum 1-A, yields essentially the same results as the original BEACON, and is an NRC approved methodology.

BEACON applies fixed uncertainty in the transient power distribution vice the variable uncertainty approved in the original BEACON topical. This 10 CFR 50.59 evaluation demonstrates the fixed uncertainty always bounds the variable uncertainty.

BEACON determines the limiting transient power distribution by using the current core model to generate the best estimate Hot Full Power ARO steady state measured power distribution. The Wz factors are then applied to determine limiting transient peak powers. This 10 CFR 50.59 evaluation demonstrates this method of determining limiting transient peak powers is consistent with the original BEACON topical.