Wolf Creek
Nuclear Operating Corporation

Ron Benham
Director Nuclear and Regulatory Affairs

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RA 21-0014

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

Commissioners and Staff:

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, 2019, to December 31, 2020, and contains a summary of 50.59 evaluations implemented 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) 364-4204.

Sincerely,

Ron Benham

Ron Benham

RDB/rlt

Attachment: Wolf Creek Generating Station Biennial 50.59 Evaluation Report

cc: S. S. Lee (NRC), w/a
S. A. Morris (NRC), w/a
N. O'Keefe (NRC), w/a
Senior Resident Inspector (NRC), w/a

WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station

Docket No: 50.482
Facility Operating License No. NPF-42

BIENNIAL 50.59 EVALUATION REPORT

Report No.: 27

Reporting Period: January 1, 2019, through December 31, 2020

SUMMARY

This report provides a brief description of changes, tests, and experiments implemented at Wolf Creek Generating Station (WCGS) 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 Plant Safety Review Committee (PSRC) for the period beginning January 1, 2019 and ending December 31, 2020. This report is submitted in accordance with the requirements of 10 CFR 50.59(d)(2).

Based on these evaluations of changes:

- There is no more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the Updated Final Safety Analysis Report (USAR).
- There is no more 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 no more than a minimal increase in the consequences of an accident previously evaluated in the USAR.
- There is no more 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 an 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 2019-0001 Revision 0

Title: 7300 Controls Upgrade

Activity Description:

The proposed activity upgrades multiple non-safety-related nuclear steam supply system (NSSS) control systems by incorporating those individual control systems into an Ovation-based distributed control system (DCS) to replace the existing Westinghouse 7300-series process control equipment. Various control system changes reflecting improvements and lessons learned from operating experience with the existing control systems are also incorporated into the new design. In addition, the proposed activity includes upgrades to the existing Ovation-based main turbine control system and feedwater pump protection and control system and integrates these systems with the NSSS control systems into the DCS.

The activity encompasses the following engineering changes:

- Design Change Package (DCP) 13324 replaces the 7300-series equipment in the NSSS control cabinets and makes changes to equipment used in the main turbine control system and the main feedwater pump protection and speed control system.
- DCP 20009 modifies the main control board (MCB) in support of DCP 13324.

The proposed activity is limited to systems and components used for plant control systems. Devices used in the 7300-series protection systems are not modified.

DCP 13324 and supporting change packages involve the instrument loops for numerous systems, with a wide range in the level of complexity of the changes. Appendix H.1 to DCP 13324 provides a description of the changes and their impact, organized on a system-by-system basis as follows:

1. Distributed Control System (DCS)
2. Reactor Coolant System (RCS)
3. Chemical and Volume Control System (CVCS)
4. Main Steam (MS)
5. Main Feedwater (FW)
6. Main Turbine (MT)
7. Residual Heat Removal (RHR)
8. Safety Injection (SI)
9. Auxiliary Feedwater (AFS)
10. Nuclear Instrumentation System (NIS)

At present, control actions and logic within the 7300-based NSSS control systems are implemented via individual analog-based cards containing discrete and integrated circuit components. Each card implements a single function or single set of related functions, although that function may be repeated multiple times on a given card. These cards are physically wired together to implement individual instrument loops, including logic, control, indication, and annunciation.

The proposed change incorporates the NSSS control systems into an Ovation-based DCS which is comprised of controllers, input/output modules, power supplies, workstations, operator touchscreens, small loop interface modules (which replace certain existing control board manual/automatic stations), switching devices, and other related equipment – the overall network configuration is similar to that currently used for main turbine and main feedwater pump control and protection. Multiple functions are performed in each controller, and all processing is performed in the digital domain under software control.

In addition to moving the individual control systems and instrument loops from the 7300-series platform to the ovation-based DCS, the proposed activity involves enhancements or changes to the major control systems, including changes to control strategies, algorithms, and the use of additional or different input parameters for control. Most of the control systems within the scope of the proposed activity are minor control systems and instrument indication loops where functional changes are minimal.

The proposed activity includes changes to the human-machine interface. Operator touchscreens will be used for indication and control in numerous instrument loops, and existing equipment that is no longer necessary is being removed from the main control board. In addition, there are other minor changes to the overall main control board layout to accommodate the new equipment.

The proposed activity is being implemented because the existing 7300-series NSSS control systems rely on original plant equipment which is experiencing increasing problems with maintenance and obsolescence. The Ovation-based DCS is a state-of-the-art digital control system which has been successfully implemented at other nuclear plants. The functional changes to individual control systems are based on WCNOC and Westinghouse experience with the control systems since the original plant design.

50.59 Evaluation:

The proposed activity upgrades multiple NSSS control systems by incorporating those individual control systems into an Ovation-based distributed control system. Various control system changes reflecting improvements and lessons learned from operating experience with the existing control systems are also incorporated into the new design. Therefore, the proposed activity consists of both a digital upgrade and functional changes to the affected control systems.

The control systems within the scope of the proposed activity include the following major control systems:

- Distributed control system
- Reactor coolant (reactor (T_{avg}) control, pressurizer pressure control, pressurizer level control)
- Chemical and volume control (reactor makeup control)
- Main steam (steam dump control)
- Main feedwater (steam generator level control, main feedwater pump speed control)
- Main turbine (main turbine control)

Failures in these major control systems could initiate the following Condition II events previously evaluated in the USAR.

- Feedwater system malfunctions that result in an increase in feedwater flow (15.1.2)
- Excessive increase in secondary steam flow (15.1.3)
- Turbine trip (15.2.3)
- Loss of normal feedwater flow (15.2.7)
- Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (15.4.1)

- Uncontrolled rod cluster control assembly bank withdrawal at power (15.4.2)
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (15.4.6)
- Chemical and volume control system malfunction that increases reactor coolant inventory (15.5.2)
- Inadvertent opening of a pressurizer relief valve (15.6.1)

Since an occurrence of any of the accidents identified above as a result of the proposed activity would be due to a failure in the modified systems, the modified systems must exhibit a low likelihood of failure so that there is no more than a minimal increase in the frequency of occurrence of these accidents. Likewise, the modified systems must exhibit a low likelihood of failure so that there is no more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

In accordance with the guidance of Nuclear Energy Institute (NEI) 01-01, "Guideline on Licensing Digital Upgrades," and Regulatory Issue Summary (RIS) 2002-22, Supplement 1, a qualitative assessment of the digital upgrade aspect of the proposed activity was performed in order to provide reasonable assurance that there was a low likelihood of failure of the modified systems. The qualitative assessment reviewed the failure analyses performed, the design attributes of the digital upgrade, the quality of the design process used in developing the upgrade, and operating experience with similar Ovation-based upgrades.

The functional changes or enhancements to the control systems were reviewed on a system-by-system basis in Appendix H.1 of DCP 13324. This Appendix includes the following:

- a description of the existing 7300-series configuration of each control system, including the operator interface at the main control board and the failure modes and effects discussed in the USAR
- identification of the changes associated with the Ovation upgrade to the control system, including changes to the operator interface
- a review of the impact of the changes on the control system, including the impact on the operator interface and on the control system failure modes and effects

On the basis of the qualitative assessment of the digital upgrade and the review of the functional changes, it was concluded that the proposed activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR or in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the USAR.

Although failures in the major control systems can cause the previously evaluated accidents identified above, failure modes and effects analyses performed in support of the proposed activity concluded that the existing failure effects remain bounding. Therefore, it was concluded that existing radiological dose consequences remain bounding, such that the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR or in the consequences of a malfunction of an SSC important to safety previously evaluated in the USAR.

As described above, existing failures and failure modes of the major control systems could result in accidents and malfunctions previously described in the USAR. It was recognized that the incorporation of multiple control systems into a network utilizing common resources introduced the potential to create an accident or malfunction not previously considered. The results of the qualitative assessment indicated that the likelihood of a common cause failure was sufficiently low – that is, comparable to the likelihood of failures that are not considered in the USAR. On that basis, it was concluded that the proposed activity does not create a possibility for an accident of a different type or a malfunction with a different result than any previously evaluated in the USAR.

Since failures in the major control systems can cause the previously evaluated accidents identified above, and since the failure modes and effects analyses performed in support of the proposed activity determined that the existing failure effects remain bounding, the existing safety analyses, which demonstrate that design basis limits for a fission product barrier are not exceeded, remain bounding for failures in the modified control systems. Therefore, it was concluded that the proposed activity does not result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered.

Finally, no USAR-described method of evaluation is affected by the proposed activity.