WOLF CREEK
NUCLEAR OPERATING CORPORATION

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RA 15-0013

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, 2013, to December 31,2014, 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) 364-4041.

Sincerely,

Steven R. Koenig

SRK/rlt

Attachment

cc: M. L. Dapas (NRC), w/a
C. F. Lyon (NRC), w/a
N. F. O'Keefe (NRC), w/a
Senior Resident Inspector (NRC), w/a

IE47
NRR

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.: 24

Reporting Period: January 1, 2013 through December 31, 2014

SUMMARY

This report provides a brief description of changes, test, 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, 2013 and ending December 31, 2014. This report is submitted in accordance with the requirements of 10 CFR 50.59(d)(2).

On the basis of these evaluation 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 2012-0002 Revision: 0

Title: Turbine Control System (TCS) Upgrade

Activity Description:

The WCGS Turbine Control System (TCS) was upgraded by replacement of the existing General Electric Mark II Electro-Hydraulic Control (EHC) system and Emergency Trip System (ETS), with a Distributed Control System (DCS) supplied by Westinghouse. The replacement system is based upon the Ovation platform supplied by Emerson Process Management (EPM).

The main turbine control system is designed to prevent turbine overspeed in the event of a sudden loss of load using multiple levels of component redundancy and diversity. Failure of any single component will not result in a turbine rotor speed exceeding the design overspeed of 120 percent of rated speed, or 2160 RPM. The function and performance of the TCS, as described in the USAR, is not being changed, with the exception being the replacement of the existing mechanical-hydraulic overspeed trip based upon the centrifugal principle with an alternate more reliable design.

50.59 Evaluation:

The TCS is non-safety related and the modification does not change the function or performance requirements for the system as described in the USAR. The TCS upgrade does not increase any plant operating parameters that would result in increased challenges to important safety components or the frequency of any accident described in the USAR. No new interface requirements with important to safety components that function to limit the consequences of an accident are established by this upgrade.

The WCGS specific TCS upgrade Software Hazards Analysis (SHA) evaluated the TCS upgrade related system-level hazards to ensure that the results would be bounded by the results of malfunctions or accidents previously considered in the USAR.

The evaluation determined that the results of potential TCS upgrade failures are enveloped by the current USAR Chapter 15 analyses.

An SHA review of the WCGS USAR Chapter 15 events was performed to identify any new system-level hazards with regard to the digital TCS upgrade. No new system-level hazards or failure modes were identified as a result of this review.

An SHA evaluation of sub-system level software failures, including software common cause failures and cyber security events/cyber-attacks, to determine their impact on the identified system-level hazards, concluded that potential sub-system level software failures would not lead to different types of accidents or impact plant USAR analyses.

A turbine trip evaluation determined the total system failure probability of both the existing TCS and the upgraded TCS to be essentially identical, 1.142E-6 and 1.08E-6 respectively.

Evaluation Number: 59 2012-0003 Revision: 0

Title: Replace current Number 1 Seal inserts in the four Wolf Creek Model 93A-1 Reactor Coolant Pumps (RCP's) with a modified design called the Shield Shutdown Seal (SDS)

Activity Description:

The Number 1 Seal inserts in the four Wolf Creek Model 93A-1 Reactor Coolant Pumps (RCPs) were replaced with a modified design called the Shield Shutdown Seal (SDS). The affected Systems, Structures, Components (SSC's) are the Number 1 Seal inserts in RCPs PBB01A, B, C, and D. The activity includes the preparation of design and configuration change documents, procurement, and actual installation of the SDS in Refuel 19.

The SDS integrated new features into the existing Number 1 Seal insert and is located downstream of the current film-riding face seal. A shoulder is machined into the inner diameter at the top flange and bore machined into the groove diameter above the shoulder. SDS sealing rings and a thermal actuator are then placed into the shoulder and bore respectively.

This activity was performed to reduce the impact of loss of all RCP seal cooling which is most likely to occur during loss of all on-site and off-site AC power (Station Blackout (SBO)). This improves margin to a severe core damage event. SBO is the dominant core damage event for WCGS.

50.59 Evaluation:

The SDS is designed to only deploy on a stationary RCP shaft and only after all seal cooling has been lost and the seal temperature rises to a prescribed temperature. The SDS will actuate when a wax material within the actuator assembly melts. Actuators are factory-tested to verify repeatable operation within their design temperature range (250 degrees F to 300 degrees F). With WCGS at 100 percent power, the number 1 seal temperatures indicate 130 degrees F.

The frequency of SDS inadvertent actuation is 1.2 E-06 events per pump-year of operation or 4.8 E-06 per year for a four-loop plant. This is a very low probability of occurrence; thus it is concluded that inadvertent actuation of an SDS during normal plant operation has negligible contribution to the overall frequency of a forced shutdown of the plant from all causes which is about 1 E-01 per year including forced outages due to RCP issues which is about 6 E-02 per year. Thus, the frequency of this previously evaluated event is not increased due to SDS installation.

The evaluation concluded that no new accident is created or any existing analyzed accident is made worse, including the consequences. There are no new, unanalyzed failures introduced because of this change in design of the number 1 RCP seal insert.

Evaluation Number: 59 2012-0004 Revision: 0

Title: Steam Generator Feed Pump Protection and Control System Upgrade

Activity Description:

The existing WCGS Steam Generator Feed Pump (SGFP) protection and control of the General Electric MDT-20 system and electro-hydraulic controls was upgraded with a Distributed Control System (DCS) supplied by Westinghouse. The digital replacement system is based upon the Ovation platform supplied by Emerson Process Management (EPM). The normal function of the SGFP protection and control system is to generate position signals for the High Pressure and Low Pressure control valves, the SGFP recirculation valves, and the condensate pump recirculation valves. Changing the position of the steam valves provides the method of controlling the SGFP turbine speed. Using the system, the SGFP turbines are capable of operation from a shutdown state to full load.

The function and performance of the SGFP protection and control system as described in the USAR, is not being changed, with the exception being the elimination of the existing electrical overspeed trip valves, associated test solenoid valves, and the mechanical overspeed trip elimination. The SGFP protection and control system upgrade eliminates existing Single Point Vulnerabilities.

50.59 Evaluation:

The SGFP protection and control system is non-safety related and the modification does not change the function or performance requirements for the system. The SGFP protection and control system upgrade does not change any plant operating parameters that would result in increased challenges to important safety components or the frequency of any accident described in the USAR. Furthermore, no new interface requirements with important to safety components that function to limit the consequences of an accident are established by this upgrade.

The removal of the mechanical overspeed trip mechanism and the electrical overspeed trip valves has no impact on accident mitigation or the consequences of an accident.

The function of tripping the SGFPs is not part of the primary success path for accident mitigation and does not impact any USAR transient or accident analyses.

The WCGS specific SGFP protection and control system upgrade Software Hazards Analysis (SHA) evaluated the SGFP protection and control system upgrade related system-level hazards and concluded that the results would be bounded by the results of malfunctions or accidents previously considered in the USAR.

The evaluation determined that the results of potential SGFP protection and control system upgrade failures are enveloped by the current USAR Chapter 15 analyses and no new system-level hazards or failure modes were identified as a result of this review.

An SHA evaluation of sub-system level software failures, including software common cause failures and cyber security events/cyber-attacks, to determine their impact on the identified system-level hazards, concluded that potential sub-system level software failures are bounded by or do not impact the USAR accident analyses previously considered.

Evaluation Number: 59 2013-0001 Revision: 0

Title: Turbine-Driven Auxiliary Feedwater Pump Controls Replacement

Activity Description:

The analog governor controls for the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) are obsolete. This change replaced the equipment with a new digital platform for the turbine governor controls and replaced the Trip and Throttle (T&T) Valve (FC HV 0312) starter controls.

This change consists of the following:

Replacement of the local control panel with new Original Equipment Manufacturer supplied controls based on a digital platform for governor controls and new motor starter for the DC motor operator for the T&T Valve.

The Operator Manual/Auto stations on the Main Control Board and the Auxiliary Shutdown Panel (RP118B) will be replaced with new simplified stations compatible with the new controls. The electrohydraulic actuator for the governor valve will be replaced with a new electric actuator.

The governor valve linkage arm and bracket will be eliminated with a new actuator bracket, aligning the actuator to a direct linear arrangement with the governor valve stem.
Replacement of the Foxboro Spec 200 control loops that supported the old Manual/Auto stations and speed setpoint and provided control signal isolation for Control Room Fire Safe Shutdown with a simple isolation device for the 24VDC circuit in the Main Control Board.

50.59 Evaluation:

The impact on the Auxiliary Feedwater System and the accidents listed in the USAR Chapter 15 were reviewed. The change to the plant and the implementation activities are bounded by the current WCGS accident and failure analyses.

There is no adverse impact to the provisions to prevent the accumulation of condensate in the turbine steam supply lines and no lessening of the system's ability to operate with steam inlet pressures ranging from 92 to 1,290 psia, as defined in the original specification M-021 and verified in the original turbine qualification testing contained in M-021-00087.

The design continues to meet the requirements defined for the exhaust line to remain functional after a Safe Shutdown Earthquake and or to perform its intended function following a postulated hazard, such as internal missile or pipe break.

Evaluation Number: 59 2013-0002 **Revision: 0**

Title: HKE25 Conveyer Car Operation with Removal of the Hold Down Latch

Activity Description:

The hold down latch on the fuel handling conveyer car was removed as the hold down latch was not engaging all the time on the reactor end of the system. The fuel bundle needs to be placed in the horizontal position to allow for transport of the fuel bundle to the spent fuel pool (SFP). The hold down latch is a mechanical latch device on the transfer car that is opened by the car moving into position in the horizontal position. The conveyer car was moving into position but the latch was not latching when loaded with a bundle and ready to return to the fuel building from the upender pit in the reactor building. This change removed the hold down latch and allowed positioning the fuel bundle from the vertical to the horizontal position for transport to the SFP.

50.59 Evaluation:

The removal of the hold down latch from the conveyer car, with the compensating action of visual verification of the car in the proper position for upending, remains a reliable means for handling the fuel assemblies. The conveyer car shall be utilized in the manual mode when the hold down latch is removed. Two redundant interlocks allow a lifting arm operation only when the transfer car is at the respective end of its travel on the reactor side as well as on the fuel building side, one of these interlocks is the position switch, the other is the hold down latch that this modification is removing. The position switch is the primary device with the hold down latch being the backup. Removal of the hold down latch will remove the redundancy but this feature will be compensated for by visual examination and verification that the car is at its proper position for upending the fuel by utilizing the positions arrows.

The removal of the hold down latch and the implementation activities are bounded by the current WCGS accident and failure analyses.

Evaluation Number: 59 2013-0002 Revision: 1

Title: HKE25 Conveyer Car Operation with Removal of the Hold Down Latch

Activity Description:

The hold down latch on the fuel handling conveyer car was removed as the hold down latch was not engaging all the time on the reactor end of the system. The fuel bundle needs to be placed in the horizontal position to allow for transport of the fuel bundle to either the spent fuel pool (SFP) or the refueling pool. The hold down latch is a mechanical latch device on the transfer car that is opened by the car moving into position horizontally. The conveyer car was moving into position but the latch was not actuating, preventing the fuel bundle from pivoting from the horizontal to the vertical position. Removing the hold down latch allowed for upending the fuel bundle for transporting it to the SFP or refueling pool. Revision 0 only allowed fuel movement to the SFP, while revision 1 also allows fuel movement to the refueling pool.

50.59 Evaluation:

The removal of the hold down latch from the conveyer car, with the compensating action of visual verification of the car in the proper position for upending, provides a reliable means for handling the fuel assemblies. The conveyer car shall be utilized in the manual mode when the hold down latch is removed. Two redundant interlocks allow a lifting arm operation only when the transfer car is at the respective end of its travel on the reactor side as well as on the fuel building side, one of these interlocks is the position switch, and the other is the hold down latch that this modification is removing. The position switch is the primary device with the hold down latch being the backup. Removal of the hold down latch will remove the redundancy but this feature will be compensated for by visual examination and verification that the car is at its proper position for upending the fuel by utilizing the positions arrows.

The removal of the hold down latch and the implementation activities are bounded by the current WCGS accident and failure analyses.

Evaluation Number: 59 2013-0003 Revision: 0

Title: Probable Maximum Precipitation (PMP) Flood Calculations

Activity Description:

WCGS Flood Calculation XX-C-023 is the Analysis of Record for determining the maximum flood elevation at WCGS. This calculation was updated and replaced by 3 new flood calculations as listed below: The new flood calculations were developed using the guidance provided in NUREG/CR – 7046 so it is compatible with WCGS's Fukushima response to perform a flooding hazard re-evaluation for the site.

Calculation 69461-C-001: determines the Probable Maximum Precipitation (PMP) using procedures outlined in Hydrometeorological Report No. 52 for WCGS.

Calculation 69461-C-002: determines the peak discharge associated with the PMP that was calculated in Calculation No. 69461-C-001 using procedures outlined in Hydrometeorological Report No. 52 for the WCGS. This calculation uses the data developed within the Calculation No. 69461-C-001 as input to the computer program Hydrologic Engineering Center – Hydrologic Modeling System (HEC-HMS) version 3.5.

Calculation 69461-C-003: estimates water surface elevations at WCGS for the PMP event. The peak discharges determined from Calculation No. 69461-C-002 are used as input into the USACE Hydrologic Engineering Center River Analysis System (HEC-RAS) software to determine the water surface elevations throughout the site.

50.59 Evaluation:

NUREG/CR-7046 requires that the water surface elevation near a safety-related building produced by the PMP be less than the floor elevation. While the calculated water surface elevations varied across the site, modeling results indicated that all safety-related building location flood stages are less than the floor elevation using the methodology and guidance provided in NUREG/CR – 7046.

The new calculations use methods that have been approved by the NRC for the intended application (NUREG/CR-7046). Additionally, they do not change how WCGS uses the flood information to operate or control SSCs to prevent flood events from adversely affecting Safety Related Seismic Category 1 Structures. The results from the new flood calculations are similar to the existing calculation of record (XX-C-023) except these provide more site-specific information. Compared with the existing flood calculation the new calculations slightly reduce margin at the north end of the powerblock and improve margin to the south, and are considered conservative or essentially the same and continue to meet the acceptance criteria.

Evaluation Number: 59 2014-0001 Revision: 0

Title: Thermocouple/Core Cooling Monitor Upgrade

Activity Description:

The Thermocouple/Core Cooling Monitor (TCCM) combines the functions of monitoring for excessive core exit thermocouple temperatures and monitoring both core exit thermocouple temperatures and hot and cold leg temperatures for saturation margin. The TCCM System has experienced obsolescence and reliability issues that have caused system components and modules to be degraded or fail.

The WCGS TCCM upgrade involved replacement of the existing microprocessor-based system supplied by Westinghouse. The replacement system is based upon the Advanced Logic System (ALS) supplied by Westinghouse. The ALS platform was installed at WCGS as a replacement of Main Steam and Feedwater Isolation System (MSFIS) equipment.

An ALS Service Unit (ASU) is allocated for each train of the TCCM and resides entirely within the TCCM cabinet. The ASU consists of a PC Node Box and a Flat Panel Display. ASU displays core temperatures, pressures, channel calibration, system status and enables the performance of system test and calibration activities. The implementation of the ASU is based on the Common Qualified (Common Q) Platform defined in Westinghouse WCAP-16097-P-A, "Common Qualified Platform Topical Report."

50.59 Evaluation:

NUREG-0737, "Clarification of TMI Action Plan Requirements," specifies the functional requirements for the TCCM. WCGS USAR Chapter 18, section 18.2.13 describes how the TCCM complies with NUREG-0737 requirements. The function and performance of the TCCM as described in the USAR, is not being changed.

WCGS USAR Chapter 7, Table 7A-3, addresses TCCM compliance with Regulatory Guide 1.97 requirements for post-accident monitoring indication. No changes are required for Regulatory Guide 1.97 compliance as described in the Table 7A-3 data sheets as a result of the TCCM upgrade because exactly the same functions in the original TCCM are replicated in the replacement TCCM.

Each train of the ALS platform-based TCCM includes an ASU. The ASU consists of a PC Node Box and a Flat Panel Display. The implementation of the ASU is based on the Common Q Platform described in WCAP-16097-P-A, "Common Qualified Platform Topical Report." The Common Q platform is a set of commercial-grade hardware and previously developed software components dedicated and qualified for use in nuclear power plants. The NRC has reviewed and approved the Common Q system for installation in safety related applications. WCAP-16097-P-A includes the NRC SER for the Common Q platform.

The built-in diversity features provided by the ALS platform significantly reduces the potential for common cause failures in TCCM functions as compared to the existing microprocessor based TCCM.

Evaluation Number: 59 2014-0002 Revision: 0

Title: ESW Below Ground Plant Tie-In Approval and Component Abandonment

Activity Description:

The Essential Service Water (ESW) system has experienced localized degradation in buried piping in the form of tuberculation, pitting, and through-wall leakage. WCGS replaced and redesigned the below ground ESW piping.

The buried ESW Train "A" and "B" supply and return piping including the warming lines and access vaults were replaced and rerouted. The supply and return piping was routed to the new circulating water crossing and required seismic support to alleviate potential concern associated with a circulating water line break washing out bedding below the new ESW. The return piping was rerouted to a new discharge point located in the ultimate heat sink (UHS). The change abandoned the original discharge structure. The warming lines were rerouted from the return piping through penetrations into the ESW pumphouse forebays. New access vaults were installed periodically along the new ESW system piping route.

This ESW system change involved revising the current tornado-generated missile protection and freeze protection configuration and seismic support configuration for portions of the ESW below ground piping and components.

50.59 Evaluation:

The change to the ESW missile and freeze protection configuration, seismic support configuration, and new discharge structure will continue to provide protection against the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles. The changes do not reduce the capabilities of the ESW system in performance of the functions described in the USAR and the system and equipment will continue to meet all of their existing safety design bases. The failure modes and effects analysis for the ESW system is not affected by the change.

The USAR was reviewed to identify the accidents/transients previously evaluated that are potentially affected by the change to ESW piping and the seismic support configuration. They are the steam generator tube rupture (SGTR) and the Loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary. The Station Blackout (SBO) analysis was also reviewed.

The new ESW missile and freeze protection configuration, seismic support configuration, and new discharge structure provides all the capabilities and functions present in the existing configuration and will not create any additional failure modes. After implementation, the ESW system continues to be capable of supplying the minimum flow rates required to remove heat from the containment and necessary safety-related components from a postulated design basis accident (DBA) and dissipate it to the ultimate heat sink, including the SBO coping time of four hours. The ESW system will continue to be capable of removing heat from plant components necessary to achieve and maintain post-fire or post-accident safe shutdown.

The ESW system will continue to be available for the mitigation of the effects associated with the limiting DBAs and transients previously evaluated in the USAR. Therefore, all assumptions utilized within USAR-described dose analyses remain valid and bounding.