Nebraska Public Power District

Always there when you need us

NLS2006084
October 12, 2006

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: 10 CFR 50.59(d)(2) Summary Report
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The purpose of this letter is to provide the summary report of evaluations that have been performed, in accordance with the requirements of 10 CFR 50.59(d)(2). This report covers the time period from August 1, 2004, to July 31, 2006. Summaries of applicable facility changes are attached. There were no changes to procedures, tests, or experiments implemented during this period that require reporting under the provisions of 10 CFR 50.59.

In accordance with 10 CFR 50.4, the original report is enclosed for your use, and copies are being transmitted to the Nuclear Regulatory Commission (NRC) Regional Office and the NRC Resident Inspector for Cooper Nuclear Station.

Should you have any questions concerning this matter, please contact me at (402) 825-2774.

Sincerely,

Paul V. Fleming
Licensing Manager

/wrv
Attachment

cc: Regional Administrator, w/attachment
USNRC - Region IV

Cooper Project Manager, w/attachment
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector, w/attachment
USNRC - CNS

NPG Distribution, w/o attachment

CNS Records, w/attachment

COOPER NUCLEAR STATION
P.O. Box 98 / Brownville, NE 68321-0098
***Telephone:** (402) 825-3811 / **Fax:** (402) 825-5211*
www.nppd.com

IE47

ATTACHMENT

FACILITY CHANGES

CED 1999-0072 Change Notice 48
(Evaluation 2002-0005 Revision 4)

TITLE: Optimum Water Chemistry (OWC) Modification

DESCRIPTION: Change Evaluation Document (CED) 1999-0072 was reported in the last 10 CFR 50.59 Summary Report. The elements of Change Notice 48 to this CED that affected previously reported Revision 3 of Evaluation 2002-0005 pertained to changes to the oxygen compression/injection functions and the injection of plant air into the Offgas System. Specifically, this activity established the permanent deletion of the oxygen injection into the Offgas System from the OWC System and the abandonment of the OWC oxygen compression skid. Instead, Plant Air is provided as an oxygen source to ensure the hydrogen concentration downstream of the Augmented Offgas recombiner skid will still be maintained below explosive limits during OWC hydrogen injection operation.

10 CFR 50.59
EVALUATION: (This 10 CFR 50.59 Evaluation is the same as that reported in the previous 10 CFR 50.59 Summary Report, except where changes have been made to reflect Change Notice 48 as noted by revision bars). OWC is a proven technology for the prevention of Inter-Granular Stress Corrosion Cracking (IGSCC) crack initiation and growth in the Reactor Coolant System. Prevention of IGSCC reduces the possibility of a large or small break Loss-of-Coolant Accident. The injection of hydrogen into the Condensate System and the injection of air into the Offgas System do not adversely affect the Condensate System, Feedwater System, Reactor Recirculation System, Offgas System or the vessel and vessel components. There is no increase in the probability of Reactor Recirculation System Pump seizure. The OWC System also injects oxygen into the condensate system, maintaining the dissolved oxygen concentration within the fuel warranty limits, thus ensuring the integrity of the fuel bundles is not adversely impacted. There are no adverse impacts on the Control Rod Drive Mechanism. Accordingly, the probability of a Control Rod Drop Accident (CRDA) is not increased. Oxygen concentration in the reactor coolant system will remain sufficiently high to prevent flow accelerated corrosion; therefore, the probability of a Main Steam Line Break Accident is not increased. The OWC does not interface with any essential structure, system, or component with the exception of the control and indication circuits; therefore, the installation of this system does not impact accident initiators, and will not increase the possibility of an accident previously evaluated.

The OWC does not adversely affect any equipment or affect accident mitigation assumptions. The increase in main steam line radiation level is controlled by the test procedure to remain below the main steam line radiation monitor (MSLRM) High-High alarm setpoints, which are based on the CRDA. The Cooper Nuclear Station containment is inerted during operation and operators have the capability to manually provide makeup nitrogen. Further, the hydrogen generation and injection systems are designed to automatically trip on reactor low power, and loss of power. Evaluation has shown that hydrogen injection for up to 60 minutes following a reactor scram is acceptable. Therefore, the consequences of an accident are not increased.

Design provisions minimize the possibility for hydrogen leakage in the system and generation equipment is located in a dedicated building. Hydrogen monitors are located in areas where leakage is possible to alert operators should leakage occur. Evaluation of these areas has determined that any leakage would disperse and not concentrate. Therefore the risk of a fire or explosion has not been increased.

The MSLRM High alarm setpoint has been raised due to increases in background radiation levels as a result of hydrogen injection. However, an Environmental Qualification (EQ) evaluation demonstrated that when actual hydrogen gas injection into the plant occurs, the EQ equipment can withstand the estimated increase in background radiation levels for the remainder of plant life. Additionally, the system has been designed and evaluated to minimize the impact of an excessive hydrogen flow rate transient.

The OWC Injection System final phase of testing injects hydrogen and plant air (during normal operation and on system shutdown) into the plant in a slow controlled manner. Plant chemistry and radiation level responses are closely monitored during hydrogen injection increases, decreases, and trips. Plant procedures have been revised to ensure that the OWC Injection and Gas Generation Systems are placed in a safe condition if other plant events command the attention of the Control Room operators. Hydrogen injection is also secured upon receipt of an MSLRM High alarm. Provisions are made in the design for a normal and emergency system shutdown from the Control Room.

For the reasons stated above, neither the injection of hydrogen and oxygen into the Condensate System and air into the Offgas System, nor the interfaces of the OWC System with plant equipment will increase the probability or consequences of equipment malfunction, nor create the possibility of a new accident or malfunction, or affect any Design Basis Limits for Fission Product Barriers or methodologies described in the Updated Safety Analysis Report.

EE01-147, Revision 1
(Evaluation 2003-0008)

TITLE: Summary of Main Steam Isolation Valve (MSIV) Leakage Pathway to the Condenser Seismic Qualification

DESCRIPTION: CNS License Condition 2.C.(6) required the Nebraska Public Power District (NPPD) to submit to the Nuclear Regulatory Commission (NRC) a seismic evaluation of the MSIV leakage pathway, the main turbine condenser, and the turbine building. The purpose was to obtain NRC agreement that these non-safety-related structures, systems, and components were seismically robust and would withstand the loadings of a Safe Shutdown Earthquake (SSE), and could therefore be credited for Loss-of-Coolant Accident (LOCA) dose consequence mitigation. Despite the need for NRC approval of the seismic review in order to obtain credit in the LOCA analysis, the actual development and implementation of the associated Engineering Evaluation (EE) was performed under the provisions of 10 CFR 50.59.

The purpose of this EE was to: 1) provide a summary of the activities completed to evaluate the MSIV leakage pathway to the condenser seismic qualification in order for NPPD to meet License Condition 2.C.(6), 2) to document acceptance of Stevenson and Associates report AR-001, Rev. 0, "Seismic Evaluation of MSIV Leakage Pathway at Cooper Nuclear Station," and 3) provide design control authorization for calculations prepared to support the seismic qualification evaluations.

Revision 1 of EE01-147 incorporated information from NPPD's response to Requests for Additional Information issued by the NRC and updated the status of modifications completed under CED 6007261 to resolve the outliers identified in Revision 0 of the EE. This revision also implemented the commitment made to the NRC under NLS2002120 dated 9/27/02 to revise this EE. This included the development of a new Turbine Building and Reactor Building Floor Response Spectra (FRS) for use in analyzing piping systems within the scope of the pathway.

10 CFR 50.59
EVALUATION: The implementation of the seismic qualification methodology assures that an SSE will not affect the ability of the MSIV Leakage Pathway to the main condenser to perform its required functions. Therefore, neither the frequency of occurrence or consequences of an accident are more than minimally increased. Similarly, this methodology provides assurance that there is not more than a minimal increase in the likelihood that these SSCs will fail during an SSE, and that the consequences of such failures are not more than minimally increased. No new accidents are postulated to be created as a result

of implementing this EE. Any malfunctions of the applicable SSCs within the scope of this EE have the same results as previously evaluated in the Updated Safety Analysis Report. No fission product barrier is associated with or affected by this EE. The seismic qualification methodology comports with the NRC approved Boiling Water Reactor Owners Group (BWROG) Topical Report NEDC-31858P, "BWROG Report for Increasing MSIV Leakage Rates Limits and Elimination of Leakage Control Systems." The development of the Turbine Building and Reactor Building FRS was consistent with Standard Review Plan guidance and has been accepted by the NRC as documented in Safety Evaluations in the licensing of other nuclear plants (e.g. WNP-2). Accordingly, the proposed activity is not a departure from an accepted method of evaluation.

EE04-063
(Evaluation 2004-0005)

TITLE: Evaluation of Control Room Habitability Hazardous Chemical Analysis and Assessment

DESCRIPTION: EE04-063 implemented the latest toxic gas assessment on Control Room habitability. This was based on the latest available hazardous material information and changes in potential industrial and transportation hazards located within 5 miles of Cooper Nuclear Station (CNS). The toxic hazards assessment was based on the methodology discussed in Regulatory Guide 1.78, Revision 1. This assessment was performed in support of the CNS response to Generic Letter 2003-01.

10 CFR 50.59
EVALUATION: This activity establishes an updated Control Room habitability study. The Engineering Evaluation (EE) and its associated documentation do not introduce new accident initiators or new failure modes, therefore this activity does not increase the frequency of occurrence of an accident previously evaluated in the Updated Safety Analysis Report (USAR). The activity does not affect the physical functions of structures, systems, or components (SSCs), and the gas concentrations have been evaluated as within toxicity limits to the Control Room operators. Therefore, there is no increased likelihood of a malfunction of an SSC important to safety. The revised habitability study has no increase in the consequences of an accident previously evaluated in the USAR. Since no new malfunction initiators or failure modes to SSCs are introduced, the EE does not result in an increase in consequences of an SSC malfunction previously evaluated in the USAR. Similarly, no new possibilities are created for a malfunction of an SSC important to safety with a different result than previously evaluated in the USAR. This EE does not

change the existing USAR conclusion that the Control Room is habitable under potential toxic chemical releases due to an industrial accident. Therefore, the EE does not result in a new accident of a different type than previously analyzed in the USAR. This EE has no impact on any fission product barrier. This activity utilizes Regulatory Guide 1.78, Revision 1. This results in a new computer code HABIT which replaces VAPOR as the existing code of record. The results generated by HABIT are acceptable because the code is endorsed by the Nuclear Regulatory Commission for Control Room habitability evaluations. Accordingly, there is no departure from a method of evaluation.

CED 6014563
(Evaluation 2004-0009)

TITLE: Replacement/Upgrade Main Turbine Low Pressure Rotors

DESCRIPTION: The two Westinghouse Low Pressure (LP) Turbines, rotors, blades, and other related steam path components, were replaced by two new units designed by Siemens Westinghouse Power Corporation (SWPC). The new LP turbines were of an improved design over the previous turbines, with improvements in materials, steam path design, and blade design. The specific elements of this design change that required a 10 CFR 50.59 Evaluation were:

- An updated missile analysis for the turbine overspeed event up to 120% of rated turbine speed utilizing SWPC Technical Report S32M7_10409. The SWPC technical report used a deterministic approach that demonstrated that turbine missiles will not escape the LP turbine casing.

- The new heat balance supplied by SWPC resulted in a developed feedwater temperature that is lower than the previous design feedwater temperature. Feedwater temperature is an input in various Cooper Nuclear Station (CNS) accident and transient analyses. The General Electric evaluation concluded that there is a negligible impact to the affected analyses of record.

10 CFR 50.59
EVALUATION: The Main Turbine is not an accident initiator in the USAR. Additionally, the SWPC Technical Report demonstrates that turbine missiles will not exit the turbine casing when released at or below 120% of turbine rated speed. Therefore, there is not an increase in the frequency or consequences of an accident previously evaluated in the Updated Safety Analysis Report (USAR). The new LP Turbine design has been proven in operation at other operating nuclear plants, and has no adverse effect on the seismically rugged Main

Steam Isolation Valve leakage pathway credited in the CNS Loss-of-Coolant Accident analysis. Accordingly, there is not more than a minimal increase in the likelihood of occurrence or consequences of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the USAR. The new LP Turbine does not introduce any new failure modes, and the operational functions of the Main Turbine will remain the same as with the existing turbine. Accordingly, the new LP Turbine rotors do not create a possibility for an accident of a different type than any previously evaluated in the USAR. The operation of the turbine and its subsystems will remain the same, and will be within the design and operational boundaries of the equipment and connected systems. Accordingly, the LP Turbine rotor replacement does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR. The LP Turbine rotor replacement has no effect on any fission product barrier. The revised turbine missile analysis uses the same deterministic methodology as the existing analysis. Accordingly, there is no departure from a method of evaluation described in the USAR used in establishing the design basis or in the safety analyses.

<u>Cycle 23 Core Reload</u>
(Evaluation 2005-0001)

TITLE: Core Reload Process – Cycle 23

DESCRIPTION: The Cycle 23 reload was evaluated using the Nuclear Regulatory Commission (NRC)-approved methodology for Cooper Nuclear Station (CNS) as specified in Technical Specification 5.6.5. The reload included 164 new bundles and discharged the remaining GE9 bundles. The reload operated under the Maximum Extended Load Line Limit and Increased Core Flow operating domain. No changes in operating domain or fuel type were analyzed in conjunction with Cycle 23. NRC-approved methodologies implemented this cycle included lattice physics method TGBLA06, 3D simulator code PANAC11, and stability licensing code ODYSY.

10 CFR 50.59
EVALUATION: The proposed change will set the limits for operation of the core and fuel to ensure that it meets the requirements set forth in the Updated Safety Analysis Report (USAR). No structures, systems, or components (SSCs) are being modified and no assumptions for accidents are being changed such that there could be an increase in frequency for any accident or malfunction previously evaluated in the USAR. The change calculates the limits that are required for operation such that the fuel cladding will not violate the requirements in the Technical Specifications and USAR. Use of these limits will ensure that the

consequences for accidents and malfunctions as described in the USAR will not be affected by this change. This change sets the fuel limits used during plant operation for the cycle. No physical modifications to SSCs will be made, no assumptions on SSC operation will change, and no manual actions are being substituted for automatic actions. Therefore, this change will not create the possibility of a different type of accident. No physical modifications to SSCs will be made and no assumptions on how SSCs are operated will change. This change is not introducing anything that could create the possibility for malfunction different than already assumed in the USAR. This change sets the limits to be used during the operating cycle that will ensure no design basis limits are challenged. This change is predicated on NRC approval of the Cycle 23 Safety Limit Minimum Critical Power Ratio values. Three new methodologies previously approved by the NRC for use are being adopted by CNS starting with Cycle 23. Accordingly, none of these methodologies represent a departure of evaluation described in the USAR used to establish design bases or in the safety analyses.

<u>EE05-071</u>
(Evaluation 2005-0003)

TITLE: Implementation of Calculation NEDC 02-064 to Address Establishing MSIV Leakage Pathway.

DESCRIPTION: EE05-071 implemented Change 2C2 to Calculation NEDC 02-064. This change evaluated the effects of increasing Main Steam Isolation Valve (MSIV) leakage from 11.5 scfh in one Main Steam Line to 46 scfh in the manual configuration of the MSIV Leakage Pathway, in support of a License Amendment Request. There were two credited completion times affected by this change: a) the configuration of valves accessed from the Turbine Building heater bay, and b) performance of the Turbine Stop Valve shaft alignment and closure of AS 682 and 683 from the operating floor. The heater bay calculational revision reduced some unnecessary analytical conservatisms and used MicroShield to project the maximum dose to the Station Operator(s) performing the evolution. The net effect was that the Station Operator will receive a maximum dose of 1.8 rem whole body (versus a negligible dose) during the 2 ½ hour required completion time. The operating floor calculational revision reduced the 30-hour required completion time by a factor of four (7.5 hours) to avoid excessive radiological dose.

10 CFR 50.59
EVALUATION: Revising the timing calculation for configuring the MSIV Leakage Pathway is not salient to increasing the frequency of occurrence of an accident, creating the possibility of an accident of a different type, or exceeding/altering Design

Basis Limits for Fission Product Barriers. Since the evolution will still be performed in a timely manner without undue radiological concerns and without additional concerns for valve maloperation, the proposed activity does not result in more than a minimal increase in either the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the Updated Safety Analysis Report (USAR), or the consequences of a malfunction of an SSC important to safety previously evaluated in the USAR, and does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR. Since the resulting "mission dose" to the Station Operator remains within the limits of General Design Criterion 19, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR. The activity does not constitute a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses.

Correspondence Number: NLS2006084

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None		