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NLS2012085 50.59(d)(2)
October 10, 2012

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

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District to provide the summary report of evaluations that have been performed for Cooper Nuclear Station, in accordance with the requirements of 10 CFR 50.59(d)(2). This report covers the time period from August 1, 2010, to July 31, 2012, with exceptions as described in Attachment 1. Summaries of applicable facility changes are discussed in Attachment 1. Summaries of other changes are discussed in Attachment 2. There were no 10 CFR 50.59 evaluations prepared specifically in support of procedure changes during this reporting period. There are no commitments contained in this letter or attachments.

Should you have any questions concerning this matter, please contact me at (402) 825-2904.

Sincerely,
David Nelson Madsen
for David W. Van Der Kamp

David W. Van Der Kamp
Licensing Manager

/dm

Attachments: 1. Facility Changes
2. Other Changes

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

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

Senior Resident Inspector, w/attachments
USNRC - CNS

NPG Distribution, w/o attachments

CNS Records, w/attachments

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ATTACHMENT 1

FACILITY CHANGES

With the exception of Change Evaluation Documents (CEDs) 6023100 and 6027780, the following lists those facility changes that were implemented at Cooper Nuclear Station (CNS) during the time period from August 1, 2010, to July 31, 2012.

CEDs 6023100 and 6027780 and their associated 10 CFR 50.59 Evaluations were not reported in the 2008 and 2010 10 CFR 50.59(d)(2) Summary Reports, as required; therefore, they are being reported below. The failures to report these CEDs have been entered into the CNS Corrective Action Program.

<u>CED 6023100 (Including Change Notices 1 through 5)</u>
(Evaluation 2007-0003, Revision 0)

Title: Reactor Building Crane Upgrade

Description: Due to the age of the reactor building crane's (RBC or crane) hardware, degrading reliability, and the difficulty of obtaining spare parts, the following items will be replaced with new equipment:

- Main and auxiliary hoist motors
- Bridge and trolley controllers and motors
- Main and auxiliary hoist brakes
- Bridge and trolley primary brakes

In addition the following enhancements /upgrades will also be made to the RBC:

- Load path limit switches will be replaced with a positioning programmable logic controller.
- Operator Control Panel Upgrade.
- Addition of new refuel floor radio operated control system.
- Addition of bridge and trolley tackle pull-points.
- Replacement of existing trolley bus-bars with a cable festoon system and bridge collector shoes.
- Upgrade of runway conductor bus-bar system with the addition of a dedicated ground bus-bar.
- Upgrade of cable power feed system, with new cables, increased power capacity, and a manual disconnect.

10 CFR 50.59
Evaluation: There are no accidents as described by the Updated Safety Analysis Report (USAR) that are directly or indirectly impacted by this modification. The crane itself is not discussed as an initiator either in normal operation or in any failure mode in any accident described in the USAR through its design functions or its failure to perform those design functions.

The modification to the RBC does not introduce the possibility of a change in the consequences of an accident because the RBC is not an accident initiator, nor is it required to mitigate any accidents. No new failure modes are introduced. The modifications to the RBC do not affect any other Structure, System, or Component (SSC) important to safety with any increased risk.

In conclusion, the modification of the RBC as described above and as part of this evaluation cannot cause an accident, introduce the possibility of a change in the consequences of an accident, introduce new failure modes due to their failure, nor introduce any new accident or scenario not already bounded by the safety analysis and does not revise or replace an USAR described evaluation methodology that is used in establishing the design bases or used in the safety analyses.

Therefore, this modification can be implemented without prior Nuclear Regulatory Commission (NRC) approval.

<u>CED 6029940 (Including Change Notices 1 through 12)</u>
(Evaluation 2009-022, Revision 1)

Title: Supplemental Diesel Generator

Description: CED 6029940 will modify the CNS Emergency Power System (EPS) as described in Section VIII-3 of the USAR to connect a new commercial-grade, non Safety-Related Supplemental Diesel Generator (SDG). The CED will provide the capability to isolate the Emergency Station Service Transformer (ESST) and connect the SDG to the nonsegregated bus duct that presently connects the ESST to critical 4.16kV buses 1F and 1G. The SDG is additional source of power to either bus 1F or 1G in the event of a Station Blackout. It can also power either bus 1F or 1G in the event of a Loss of Offsite Power concurrent with a failure of a single Emergency Diesel Generator. In this respect the SDG will perform a function similar to the 161kV system, the 69kV system or to the Emergency Diesel Generators.

The SDG will be periodically tested through the ESST back through the 69kV system, or a load bank connected to the SDG Bus 1S Spare Breaker. Though the SDG is not safety-related nor credited in any safety analysis, the test involving the ESST and 69kV system is not currently described or evaluated in the USAR and involves the EPS and the ESST which are described in the USAR.

10 CFR 50.59
Evaluation: There are no accidents as described by the USAR that are directly or indirectly impacted by this modification to add the SDG system. The ESST and the EPS are not an initiator of any accident or transient. The SDG system is a new system and is not discussed as an initiator either in normal operation, testing, or in any failure mode in any accident described in the USAR through its design functions or its

failure to perform those design functions. Like the EPS the SDG is not safety-related and not single failure proof.

The installation, testing, and operation of the SDG system does not introduce the possibility of a change in the consequences of an accident because the EPS system is not an initiator of any accidents. There are no SDG failure modes which would result in any different result from the current configuration. The SDG system cannot affect any other SSC important to safety with any increased risk by increasing the frequency or consequences of any malfunction.

In conclusion, the installation, testing, or failure of the SDG system as described above and as part of this evaluation cannot cause an accident, introduce the possibility of a change in the consequences of an accident, introduce new failure modes due to its failure, nor introduce any new accident or scenario not already bounded by the safety analysis.

Therefore, this modification can be implemented without prior NRC approval.

<u>CED 6020704 (Including Change Notices 1 through 7)</u>
(Evaluation 2010-014, Revision 0)

Title: Ronan Upgrade Project - Annunciator

Description: This CED replaces the Ronan Model C Annunciator System with RTP Corp I/O and a custom Human Machine Interface (HMI) as the operator interface. A Microsoft SQL database will serve as a data repository and Sequential Events recorder. The functional design of this system includes:

- Redundant remote collection of inputs
- Redundant communication paths from inputs to the Plant Data Network (PDN)
- High-availability system processing through the use of the PDN
- Diverse presentation to the Operator with lamp boxes and video touch screens
- Redundant lamp box and horn operation
- Basic existing alarm prioritization will be maintained
- Presentation of alarms will be displayed on HMI video touch screens
- Sequential Events recording

Additionally, a portion of the 'Y' PDN, associated with the Control Room Annunciator, is being implemented using the fiber optic tubes, fiber optic cables and cabinets installed by CED 6020740. These fiber pathways provide the interconnections between Zone Switches located throughout the plant and Core Switches located in the Computer Room, thus providing high speed and redundant data communications pathways to facilitate reliable data communications among the various Annunciator components.

This CED replaces the Control Room annunciator Cathode Ray Tube video displays with new Flat Panel displays. Both "CNS Control Room Human Factors Engineering Standards and Implementation Guidelines" and "NPPD Human Factors Plan," Document No. 503-8500000-77 were utilized for the development of the new Annunciator displays and equipment labeling.

10 CFR 50.59
Evaluation: The evaluation has determined the physical changes made by this modification will have no impact on the current design basis of the Annunciation system. The CED will not affect fission product barriers or USAR described evaluation methodologies; nor will the CED introduce any new accidents or malfunctions with different results. Changes in the method of control will provide an enhancement in the mitigation of consequences for the Operator. Possible malfunctions of an SSC will not have a discernible rise, thus the change is less than minimal and may be implemented without prior NRC approval.

<u>CED 6027780 (Including Change Notices 2 and 3)</u>
(Evaluation 2010-013, Revision 1)

Title: CED 6027780 Change Notice 2 (CCN #2) - Provide Evaluation of Noble Metal Solution Injected into Reactor Feedwater (RF) System. CCN #3 - Permanent Installation of On-Line NobleChem™ (OLNC) Injection Skid and Installation of Mitigation Monitoring System (MMS) Shielding

Description: Notes: CED 6027780, Installation of OLNC Injection Taps, and CED 6027780, CCN #1, did not require a 10 CFR 50.59 Evaluation.

The MMS shielding has not yet been installed under CCN #3.

CCN #2 provides the basis for the implementation of the OLNC injection. This activity involves the injection of noble metal (platinum) into the RF system in order to slow or mitigate intergrannular stress corrosion cracking (IGSCC) in the reactor vessel and attached reactor coolant system piping. The injection results in a fine layer of noble metal being deposited onto the wetted surfaces of the reactor and associated piping. The noble metal penetrates existing cracks to help slow or mitigate crack growth. Unlike previous noble metal injections performed during outages at CNS, this activity allows for the online injection of Noble Metals while the plant is > 75% reactor core flow. The OLNC application is performed no earlier than 90 days of power operation after a refueling outage to ensure an oxide layer is developed on newly inserted fuel. The online injection results in a more even distribution of noble metals throughout the system and deeper penetration of existing cracks and crevices.

CED 1999-0082 (including change notices #1 through #4), and its associated Unresolved Safety Question Evaluation (USQE) 1999-0050, previously evaluated the effects of injecting noble metal into the reactor coolant system. The evaluation reviewed effects on the reactor fuel, reactor fuel performance, reactor coolant piping, the Reactor Recirculation System, Reactor Water Clean-up System, the reactor vessel components and attached piping material interactions, reactor water chemistry, Control Rod Drive components, reaction rates of the zirconium/water reaction, hydrogen/oxygen recombination on peak cladding temperature during a Loss of Coolant Accident (LOCA), and the effect of noble metal on the long-term post-LOCA environment. The USQE concluded that the activity was safe to perform and did not result in an unreviewed safety question. The 10 CFR 50.59 Evaluation discussed below summarizes the aspects of OLNC not previously reviewed in USQE 1999-0050 as they apply to the current 10 CFR 50.59 requirements.

CCN #3 provides for the necessary changes in documentation to make the OLNC skid permanent plant equipment. CCN #3 also installs a radiation shield at the MMS rack. This shielding is provided to reduce dose to the staff when monitoring or operating components of the MMS rack. Installation of the MMS system shielding and permanent installation of the OLNC skid will be per existing plant design. CCN #3 also documents the addition of a trip function to the OLNC skid Data Acquisition System (DAS) to trip the injection pumps for a loss of communication between the pumps and the DAS.

10 CFR 50.59
Evaluation: Installation of MMS shielding, the permanent installation of the OLNC injection skid, adding a protective trip to the DAS, and the injection of noble metal compounds into the reactor vessel via the OLNC process of injection do not affect the safety of operations or the health and safety of the public.

The addition of a protective trip function to the DAS, MMS shielding installation, injection equipment installation, and the noble metal injection process have been reviewed, and it has been concluded that plant safety will not be compromised. Installation of MMS system shielding and permanent installation of the OLNC injection skid is per existing CNS USAR design criteria. The added trip function to the DAS is a vendor recommended enhancement. OLNC application is conducted during a period when the reactor is at high core flow operating conditions. During such a period, the reactor coolant system is in a high-energy state as assumed in the applicable safety analyses in the USAR. This evaluation has concluded that the frequency of occurrence or consequences of a LOCA, Control Rod Drop Accident, Main Steam Line Break Accident and Fuel Handling Accident previously evaluated in the USAR will not be more than minimally increased since OLNC does not change the initiating conditions, assumptions or methodology associated with the accident analysis. Because of the benefit of OLNC, which is expected to reduce the potential of IGSCC, the frequency of occurrence of a LOCA will be reduced after the noble metal injection. Therefore, the frequency of occurrence and the consequences of accidents previously evaluated in the USAR will not more than minimally increase. During the time-period when OLNC application is conducted, the conditions within the reactor coolant pressure boundary are such that no new scenario can be postulated that could result in fission product release. Therefore, there is no possibility of an accident of a different type than any evaluated previously in the USAR.

Because the OLNC injection skid and its DAS, MMS system shielding and OLNC injection do not create any adverse equipment interaction or operate equipment outside of its design analysis, the likelihood of occurrence or the consequences of a malfunction of a SSC important to safety previously evaluated in the USAR is not more than minimally increased, and the creation of a new malfunction of a SSC important to safety with a different result than previously evaluated in the USAR is not created during OLNC application. After the noble metal application is completed, the reactor vessel, the reactor internals and some of the associated primary pressure boundary piping and equipment potentially become better mitigated from IGSCC, and thus, have reduced probabilities of failures and/or malfunctions.

The noble metal compound does not adversely affect surfaces treated by the noble metal, nor does the noble metal application affect system performance. Fuel design and licensing basis are not changed and fuel integrity is maintained per existing program and fuel vendor requirements during normal operations and anticipated operational occurrences. Installation of MMS system shielding, permanent installation of the OLNC injection skid and the addition of a protective trip to the DAS are per existing CNS design and software criteria and the installations do not interface with fission product barriers. Therefore the design basis limits for a fission product barrier as described in the USAR are not exceeded or altered.

The noble metal compound does not adversely affect any treated equipment such as the reactor pressure vessel, piping and related pressure-retaining components. Injection of noble metal does not require change of any safety-related design or safety analysis model or result that may affect design basis limit for a fission product barrier as described in the USAR. The noble metal application process is a work activity and therefore does not require changes to methodologies described in the USAR. Installation of MMS system shielding, permanent installation of the OLNC injection skid and addition of a protective trip to the DAS are per existing CNS design and software criteria and the installations are not associated with methods of evaluation. Therefore, noble metal application will not result in a departure from a method of evaluation described in the CNS USAR in establishing the design bases or in the safety analyses.

Therefore, it is concluded that installation of MMS system shielding, permanent installation of the OLNC injection skid, addition of a protective trip to the DAS, and OLNC application can be conducted at CNS without affecting the safe operation of the plant, or the health and safety of the public. Prior NRC approval of this activity is not required.

The Noble metal compound does not adversely affect any treated equipment such as the reactor pressure vessel, piping and related pressure-retaining components.

Injection of noble metal does not require change of any safety-related design or safety analysis model or result that may affect design basis limit for a fission product barrier as described in the USAR. The noble metal application process is a work activity and therefore does not require changes to methodologies described in the USAR. Therefore, noble metal application will not result in a departure from a method of evaluation described in the CNS USAR in establishing the design bases or in the safety analyses.

Therefore, it is concluded that OLNC application can be conducted at CNS without affecting the safe operation of the plant, or the health and safety of the public. Prior NRC approval of this activity is not required.

Engineering Evaluation (EE) 10-073, R0
(Evaluation 2011-001, Revision 0)

Title: Deletion of Type C Testing of One Barrier due to Closed Loop Analysis for Nine Penetrations

Description: The proposed activity will eliminate "Type C" Appendix J testing on 13 specific components (RHR-MOV-MO18, RHR-CV-26CV/27CV, RHR-MOV-MO274A/B, RHR-MOV-MO26A/B, RHR-MOV-MO34A/B, RHR-MOV-MO39A/B, and CS-CV-18CV/19CV) on seven Residual Heat Removal (RHR) and two Core Spray (CS) containment penetrations. Elimination of the testing will save dose and resources during refueling outages. Type C tests are measured local leak rate tests which input into the total containment leakage totals (La) allowed by the Appendix J Program and Technical Specifications. A closed loop outside containment will replace a Type C tested containment isolation valve (CIV) as one of the two credited barriers against radioactive release from the Drywell. The proposed change does not alter the physical plant or the manner in which it is operated.

10 CFR 50.59
Evaluation: The design basis for satisfying Primary Containment for the two CS and seven RHR penetrations involved is that each penetration has two barriers for isolating Containment. The two credited barriers are being moved from two separate containment isolation valve barriers to one containment isolation valve barrier and a closed system. A closed system outside containment and a tested CIV outside containment satisfies the requirements of an 'other defined basis' to meet General Design Criteria 55 and 56 as discussed in ANSI N271-1976 paragraph 3.6.4 and NUREG 0800. The non-tested CIV will continue to function as a CIV and be tested as a CIV with the exception of a local leak rate test.

The frequency of an accident or the likelihood of malfunction is not increased, since deleting Appendix J leak tests (for non-credited barriers) cannot initiate an accident. Consequences of an accident are not increased since the closed loop piping is Class 2 seismic 1, isolated from outside atmosphere, designed to withstand a LOCA and temperature/pressure equal to containment design, has overpressure protection, is protected against missiles and High Energy Line Break, and is capable of maintaining system integrity. There is no increase in the consequences of a malfunction since no new failure modes for equipment are introduced. There is no possibility of an accident of a different type or malfunction of an SSC with a different result that is not addressed in the USAR, since there is no physical change or logic change to the valves and the closed loop provides the second credited barrier. No fission product barrier design basis limit is exceeded or altered since the design basis for containment is 62 psig and the closed loop piping design pressure is 125 psig or greater. In addition, the La limit (max allowed leakage) has not changed. There is no departure from the method of evaluation since the two barrier approach for isolating containment will continue and total leakage for these affected penetrations will continue to follow the guidelines endorsed by Regulatory Guide 1.163.

Based on the discussion provided above, a License Amendment is not required.

ATTACHMENT 2

OTHER CHANGES

License Basis Document Change Request 2008-008
(Evaluation 2008-015, Revision 0)

Title: Relaxation of Technical Requirement Manual (TRM) Restrictions on Mode Changes

Description: TRM Limiting Condition for Operation (TLCO) 3.0.4 will be revised to allow entry into a MODE or other specified condition in the Applicability while relying on the associated ACTIONS, provided that 1 of 3 conditions are satisfied. These conditions are: (a) the ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time, (b) there is risk assessment performed which justifies the use of TLCO 3.0.4 to change modes, or (c) an approved allowance is provided in the Specification to be entered. This is different from the current TLCO 3.0.4 in that currently a risk assessment is not allowed to justify changing modes. Conforming changes to TRM Surveillance Requirement 3.0.4 and other TLCOs are also made.

10 CFR 50.59
Evaluation: The proposed change permits the use of risk assessments to justify changing plant modes when certain SSCs have been determined inoperable during the mode of applicability.

No physical alterations to the plant and SSCs are made by this activity. No changes are being made to operating or maintenance methods and procedures. Inoperable SSCs, when changing modes of applicability, have already had their impact and are bounded by existing analyses and Technical Specifications (TS) or TRM required actions in the next mode of operation. The dose consequences of their inoperability upon entering the next mode are no different than if the equipment had become inoperable while in that next mode. Risk assessments conducted per the requirements of 10 CFR 50.65(a)(4) do not introduce a human interaction with an SSC nor require procedure training or change management. TRM and TS safety limits and LCOs related to fuel cladding, Reactor Coolant System pressure boundaries, and containment are not changed and continue to preserve the margins of safety of these barriers. The same methodologies currently used to justify safe operation in the next mode are still applicable when changing from the previous mode of operation. No new evaluation methods are proposed by this change.

For the above reasons, this proposed activity does not require a license amendment and may be implemented without prior NRC approval.