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50.59(d)(2)
72.48(d)(2)

NLS2016058
October 7, 2016

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

Subject: 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2) Summary Report
Cooper Nuclear Station, Docket No. 50-298, License No. 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) and 10 CFR 72.48(d)(2). This report covers the time period from August 1, 2014 to July 31, 2016. Summaries of applicable facility changes are discussed in Attachment 1. Summaries of applicable procedure changes are discussed in Attachment 2. Summaries of other changes are discussed in Attachment 3. There were no 72.48(d)(2) evaluations performed during the specified time period.

There are no commitments contained in this letter.

Should you have any question concerning this matter, please contact me at (402) 825-2788.

Sincerely,

Jim Shaw
Licensing Manager

/dv

Attachments: 1. Facility Changes
2. Procedure Change
3. Other Changes

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NRR

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cc: Regional Administrator w/attachments
USNRC – Region IV

Senior Resident Inspector w/attachments
USNRC – CNS

Cooper Project Manager w/attachments
USNRC – NRR Plant Licensing Branch IV-2

NPG Distribution w/o attachments

CNS Records w/attachments

Attachment 1

Facility Changes

The following list provides a summary of 50.59 evaluations that were implemented to support facility changes that were implemented at Cooper Nuclear Station (CNS) during the time period from August 1, 2014 to July 31, 2016.

Change Evaluation Document 6016581
(Evaluation 2011-05, Revision 1)

Title: Diesel Generator (DG) Voltage Regulator Upgrade

Note: As of the date of this letter, the Voltage Regulator Upgrade has only been partially implemented. Only the unit parallel switch for #1 DG has been upgraded.

Description: The DG Voltage Regulator/Exciter is being replaced and upgraded due to the existing systems nearing the end of their forty year rated lives and a desire for improved reliability. Changes include automatic return to isochronous from parallel on an emergency start as this is the desired mode of operation for an emergency run, redundant rectifier diodes prevent failure of the rectifier due to a single diode short, diode failure indication to identify rectifier failure, an externally visible control current ammeter to allow for monitoring excitation system health, changed monitoring points and settings make collected data more useful, and increased exciter capability to provide additional margin. The reason a 50.59 Evaluation is required is that the control of the DG unit parallel switches is being changed from fully manual to automatically switching to isochronous mode if in parallel when an emergency start signal occurs.

10 CFR 50.59
Evaluation: The DGs do not affect the initiators of any accidents or the frequency that accidents occur. This change does not increase the likelihood of a malfunction of the DGs because the new components are tested and qualified similarly and improvements have been made in the failure modes and effects of the DG Voltage Regulator/Exciter. The consequences of malfunctions related to this change are bounded by those already evaluated in the Updated Safety Analysis Report (USAR). This change affects a mitigating structure, system, or component, not one that can result in an accident, therefore no accidents of a different type are created by this change. The replacement equipment can only malfunction in ways that are bounded by the existing USAR evaluated DG malfunctions. This change doesn't affect any design basis limits for fission product barriers as the DG design function continues to be supported by this change. No methodology used in any plant calculations is changed as a result of this activity.

Engineering Change (EC) 6038525/USAR Change Request 2015-021
(Evaluation 2015-03, Revision 0)

Title: Reactor Equipment Cooling (REC) Flow Switch (FS) 463 Abandonment

Description: EC 6038525 will abandon-in-place non-essential REC low flow switch REC-FS-463. The flow switch actuates Control Room annunciator M-2/E-1, FPC HX LOW REC FLOW, when REC flow from the Fuel Pool Cooling heat exchangers drops below 200 gpm. As described in USAR Chapter X, Section 6.6.1, the REC non-critical loop low flow alarm switches exist to alert operators of failure in the Seismic Class IIS system piping and prompt action to isolate the non-critical header or initiate Service Water backup to REC. The low flow alarm is not required for the REC system to perform its design function of providing cooling to the Emergency Core Cooling System areas, however, the change reduces the number of alarms available to alert operators of a break in REC Class IIS piping and thus is considered an adverse effect on the method of performing the design function.

10 CFR 50.59
Evaluation: The proposed activity may be implemented without prior Nuclear Regulatory Commission approval and without obtaining a License Amendment.

The affected REC low flow alarm switch is utilized to provide indication of a potential Seismic Class II pipe failure. The switch is a passive device which signals non-essential annunciation only upon water flow decreasing below a setpoint. The switch is not an initiator of any accident, or structure, system, or component malfunction. Therefore the activity does not increase the frequency of an accident or likelihood of equipment malfunction previously evaluated in the USAR.

The elimination of flow switch REC-FS-463 will not affect the safety objective of the REC system to provide cooling to the Core Standby Cooling Systems areas. Operations uses a diverse set of reliable alarms and indication to identify REC pipe breaks and/or a loss of REC. Annunciators for REC system pressures and REC surge tank low level as well as monitoring under Procedure 6.LOG.601 will continue to provide the best indication of REC system integrity. Automatic non-critical header isolation on loss of REC pressure also provides assurance a Class IIS pipe break will not prevent operations from maintaining adequate cooling to essential equipment. Additionally, the CNS internal flooding analysis, Nuclear Engineering Design Calculation 09-102 Rev. 1, does not credit manual actions taken as a result of the subject low flow switch to mitigate flooding from a break in the associated piping. Therefore the change will not prevent or degrade the effectiveness of mitigative actions assumed in the USAR and as such the activity will not result in an increase in the consequences of an accident or equipment malfunction previously described in the USAR.

Abandoning the flow switch in the non-critical loop of REC will result in a passive appendage on the pipe. This pipe configuration is identical to the original design and any credible passive failure of the abandoned switch is bounded by the analysis of a passive REC pipe break. Therefore, the activity does not create a possibility for an accident or equipment malfunction of a different type than previously evaluated in the USAR.

This activity has no effect on any of the fission product barriers of fuel cladding, reactor coolant pressure boundary, or containment.

No input or methodology which is used in establishing the design basis or used in the safety analysis is changed with this activity.

Attachment 2

Procedure Change

The following provides a summary of a 50.59 evaluation that was implemented to support a procedure change that was implemented at Cooper Nuclear Station (CNS) during the time period from August 1, 2014 to July 31, 2016.

Procedure 2.0.1.3/Updated Safety Analysis Report (USAR) Change Request (UCR) 2015-028 (Evaluation 2015-04, Revision 0)

Title: Time Critical Operator Action for Suppression Pool Cooling (SPC)

Description: The generic Boiling Water Reactor-4 initiation time for the Residual Heat Removal (RHR) system in the SPC Mode used in the station's Anticipated Transient Without Scram (ATWS) analysis (Nuclear Engineering Design Calculation 94-034I) has proven to be unrealistic with only one minute for operators to manually align the RHR system after ten minutes of no operator action. After investigation into the sensitivity of Primary Containment temperature to this initiation timing, the demonstrated and required initiation timings are being relaxed. CNS Procedure 2.0.1.3, "Time Critical Operator Action Control and Maintenance," will be updated to reflect a more realistic time of thirty minutes for operator manual action to line up the RHR System in the SPC Mode following an ATWS. USAR Chapter XIV, Section 5.9.3, "Anticipated Transient Without Scram," will be updated to reflect the required time of 43.5 minutes (208°F Peak Suppression Pool Temperature) determined by Engineering Report 15-003. Thirty minutes is chosen for two reasons, 1) to provide margin to the 43.5 minute required time and ensure a peak suppression pool temperature less than 208°F and 2) establish a time for Procedure 2.0.1.3 that is readily achievable. The two activities are as follows:

1. Procedure Change Request to CNS Procedure 2.0.1.3, "Time Critical Operator Action Control and Maintenance"

2. UCR to USAR Chapter XIV, Section 5.9.3

10 CFR 50.59 Evaluation: This activity can be implemented without prior Nuclear Regulatory Commission approval.

Key design features credited in the ATWS analysis are Standby Liquid Control, Alternate Rod Insertion and Recirculation Pump Trip. While the initiation timing of the RHR System is a credited manual action in the station's ATWS analysis, the impact to Primary Containment temperature and pressure response remains well within the design limits of Primary Containment. This margin is the station's and the change to margin does not represent a significant reduction. The new initiation timing of the RHR system in the SPC mode in conjunction with these key design features which are unaffected assure that fuel integrity, reactor integrity, and primary containment integrity are assured under ATWS conditions.

The applicable requirements established by Nuclear Energy Institute 96-07 Revision 1, Section 4.3.2, Example 4, have been answered and show that the change to the manual action proposed by this evaluation will not result in a more than minimal increase in likelihood of any of the criteria set forth by the 10 CFR 50.59 process.

The frequency of occurrence and the consequences of accidents evaluated in the USAR are not changing due to the proposed activity.

The proposed activity does not introduce the possibility of a different type of accident, or create the possibility for a malfunction of a structure, system or component important to safety with a different result than previously evaluated in the USAR.

This activity does not result in the design limits of a fission product barrier being exceeded or changed.

Attachment 3

Other Changes

The following list provides a summary of 50.59 evaluations that were implemented to support other changes that were implemented at Cooper Nuclear Station during the time period from August 1, 2014 to July 31, 2016.

Updated Safety Analysis Report (USAR) Change Request (UCR) 2014-019
(Evaluation 2014-03, Revision 0)

Title: Torus Recoat

Description: This UCR vacates the License Renewal commitment to recoat the torus within three years of the period of extended operation (PEO). It also extends to the PEO the commitment to desludge and inspect the torus and complete analyses that demonstrate that the projected pitting of the torus will not result in reduction of torus wall thickness below minimum acceptable values. Finally, the UCR formally credits cyclic desludging, inspection, pit repair, and high water quality in maintaining the protective function of the coating system. The determination to not recoat the torus is a Nebraska Public Power District management decision that is based on the recognition that nuclear safety can be assured through the PEO without performing this activity.

10 CFR 50.59
Evaluation: Not recoating the torus and crediting other mitigating strategies to control corrosion are not precursors to any accidents described in the USAR, or of a different type not described in the USAR. The change is judged not to reduce the effectiveness of the Containment Inservice Inspection program, and thus will neither increase the likelihood of a malfunction of equipment important to safety, nor create the possibility of a structure, system or component malfunction with a different result. There is no impact on the radiological consequences of accidents or malfunction previously evaluated in the USAR. This change does not cause the Containment Pressure Design Basis Limit for the Fission Product Barrier to be exceeded or altered, and is not associated with any methodology described in the USAR.

Engineering Change (EC) 2014-012
(Evaluation 2015-01, Revision 0)

Title: Control Building Essential Ventilation System Calculation Corrections

Description: EC 14-012 addresses numerous discrepancies noted in design calculations associated with the Control Building Essential Ventilation System. During preparation of these calculations and calculation revisions, several scenarios were identified in which the Essential Ventilation System would not automatically initiate on high room temperatures. This is a concern due to hydrogen accumulation in the Battery Rooms and Reactor Protection System Rooms. Additionally, operating experience has shown that there are certain conditions that, if the Essential Ventilation System is in operation, will prevent the system from automatically stopping prior to Battery Room temperatures falling below preset values.

To address the conditions of hydrogen accumulation and low Battery Room temperatures, procedure changes and setpoint change requests are implemented via EC 14-012.

The results of a 10 CFR 50.59 screen showed that only the implementation of time constraints for operator actions to maintain Battery Room temperatures and hydrogen concentration required a 10 CFR 50.59 evaluation. Therefore, this evaluation will focus on the operator actions implemented via the procedure changes effected by EC 14-012.

For the purpose of this evaluation, the safety design bases listed in USAR Section X-10.3.2 for the Control Building Essential Ventilation System will be conservatively treated as design functions vice design features, even though they are not directly credited in USAR Chapter XIV analyses. This is conservative as it requires a more stringent evaluation to ensure the proposed activity is acceptable.

10 CFR 50.59
Evaluation: This activity can be implemented without prior Nuclear Regulatory Commission approval.

Manually starting or stopping the Control Building Essential Ventilation System is not the initiator of any accident condition; therefore, it cannot cause an increase in the frequency of occurrence of an accident.

This activity meets the requirements for addressing a new manual action that supports a design function credited in the safety analyses (Nuclear Energy Institute 96-07, Section 4.3.2, Example 4). Plant procedures and operator training programs are modified by this activity to implement the actions and demonstrate that the actions can be completed within the times required. The actions introduce minimal operator burden, can be implemented in a short period of time from the Control Room, and are contained in a normal station operating procedure. The evaluation considered the ability to recover from credible errors while performing the manual actions and the time required to recover from those errors. The evaluation also considered the effect of the change on plant systems and found it to be acceptable. Thus, the Nuclear Energy Institute 96-07, Section 4.3.2, requirements are satisfied and it has been shown that there is not a more than minimal increase in likelihood of a malfunction of a structure, system or component important to safety.

The consequences of the accidents evaluated in the USAR are not changing based upon this activity. Therefore, the activity does not result in a more than minimal increase in the consequences of an accident previously evaluated in the USAR.

The results or consequences would be no different than if the Control Building Essential Ventilation System failed to perform its function for any other reason. The consequences would not become larger; they would remain the same as before since they are not changed by this activity. Therefore, the activity does not result in a more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR.

No new types of accidents could be created with this activity. Therefore, this activity does not create a possibility for an accident of a different type than any previously evaluated in the USAR.

Manually starting or stopping the Control Building Essential Ventilation System does not create the possibility for a malfunction of any structure, system, or component. Therefore, it cannot possibly create a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the USAR.

This activity does not affect any limits related to fission product barriers.

The USAR does not describe a method of evaluation that was used to determine that the Control Building Essential Ventilation System would perform its safety design functions. Therefore, this activity does not result in a departure from a method of evaluation describe in the USAR used in establishing the design bases or in the safety analyses.