Nebraska Public Power District

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NLS2008094 50.59(d)(2)

October 14, 2008

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, 2006, to July 31, 2008. Summaries of applicable facility changes are discussed in Attachment 1. Summaries of other changes are discussed in Attachment 2. There were no changes to procedures implemented during this reporting period under the provisions of 10 CFR 50.59.

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

Sincerely,

David W. Van Der Kamp

David W. Van Der Kamp
Licensing Manager

/bk

Attachments

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

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ATTACHMENT 1

FACILITY CHANGES

Change Evaluation Documents (CED) 6010820 and 6016542
(Evaluation 2003-0009 Revision 2)

Title: Reactor Vessel Level Control System - Phase 2

Description: The modification to the reactor vessel level control (RVLC) system upgraded the single channel analog control system with a fully integrated digital control platform. The modification included removal and replacement of analog components and installation of new process transmitters, as well as changes in control board indicators and meter banding. The modification was implemented in response to spare part shortages, obsolete equipment, excessive system maintenance requirements, and control system failures that resulted in plant trips or power reductions.

Title: Reactor Feedpump Turbine Control Upgrade - Phase 2 and Phase 3

Description: This modification replaced the feedwater control system between the RVLC system and the pneumatic final drive. The modification included replacement of the control system and supporting system components including startup valve control and the turbine supervisory instruments monitoring system. The modification also added new analog signals to the upgraded RVLC system. The new capability includes modulating the minimum flow valves and providing the RVLC system with valve position and control signal input/output capability; as well as moving startup valve control to the RVLC system in addition to providing automatic differential pressure control for the feedwater startup valves.

10 CFR 50.59 Evaluation: CEDs 6010820 and 6016542 were implemented together and were evaluated under one 10 CFR 50.59 evaluation. The evaluation concluded that the upgrades made per the modifications do not affect accident initiators or system interface assumptions previously assumed in the Updated Safety Analysis Report (USAR). The existing RVLC system and reactor feedpump turbine (RFPT) speed control performance requirements are unchanged. The RFPT modification does not affect feedwater or other system piping boundaries and the RVLC system modification provides system enhancements which improve system response to an abnormal event. Therefore, the upgrades will not increase the frequency of or adversely impact the occurrence of any accidents or transients described in the USAR. The upgraded RVLC system is designed to meet seismic II/I criteria within the control room and cable spreading room.

The failure mechanism for the loss of feedwater flow or controller failure has not changed as a result of the RFPT modification; therefore, there are no significant impacts on system failure modes. Accordingly, there is no increased frequency of malfunction of equipment important to safety, nor increased consequences resulting from such failures. The upgrades will have no potential for the creation of an event of a type not previously evaluated in the USAR, nor will the modification result in the introduction of new failure modes not previously evaluated in the USAR.

There are no new methodologies associated with these modifications or challenges to design basis limits for fission product barriers. Therefore, Nuclear Regulatory Commission (NRC) approval was not required prior to implementing this change.

<u>CED 6016580 (Including Change Notices 1 through 7)</u>
Evaluation (2006-0001 Revision 0)

Title: Turbine Generator EH Fluid Automatic Temperature Control

Description: This modification installed an automatic temperature control valve in the turbine equipment cooling (TEC) system to the electro hydraulic (EH) governor coolers to maintain turbine generator fluid pump suction temperature within the required operating range. The modification also installed an EH reservoir bypass filtration/cooling skid to provide enhanced filtration and temperature control of the EH fluid supply system.

10 CFR 50.59
Evaluation: The addition of the TEC control valve and the EH skid has no impact on the plant safety analysis or the operation of plant safety systems and associated equipment. The system level potential failure modes and effects for the new equipment will not cause a different type of accident than presented in the USAR. The new equipment failure modes are considered bounded by the transient events listed in the USAR because the effect would be no worse than that previously evaluated in the USAR. Installation of the modification does not introduce a new accident initiator and does not impact or increase accident dose release or consequence. The addition of an automatic temperature control valve is a change in the mode of performing or controlling a design function for TEC cooling flow control but does not result in a new or different accident or transient previously analyzed. This change has no impact on design basis limits for fission product barriers, nor is it a change in methodology described in the USAR. Therefore, NRC approval was not required prior to implementing this change.

CED 6020580
(Evaluation 2006-0003 Revision 0)

Title: DEH Controls Jumper to Prevent Unwarranted Mode Change

Description: This modification installed a switched electrical jumper in the digital electro hydraulic (DEH) system to keep the speed/load reference signal to the DEH pressure control loop biased high during plant operation until the turbine is tripped off line. The switch in this jumper will be opened after the main turbine is tripped to support turbine generator startup activities and the switch will be re-closed after the generator is connected to the grid. This is considered a defense in depth and serves to force the existing DEH digital sub-system malfunction protective feature while the main generator is connected to the grid. Forcing this function ensures the DEH pressure controllers maintain control of the governor and bypass valves during normal operations and do not transfer to speed/load control due to component malfunctions.

10 CFR 50.59
Evaluation: The modification does not result in an increase in the frequency of occurrence of an accident previously evaluated in the USAR because the switched jumper is defense in depth only and does not introduce new failure mode effects beyond those previously evaluated. The installation of the new jumper will not initiate any new malfunctions. There is no increase in the likelihood of occurrence of a malfunction of a system, structure, or component (SSC) important to safety previously evaluated in the USAR.

The DEH system is not used to mitigate the consequences of any accidents and the new jumper will not initiate any new accidents. This modification will not impair or prevent emergency core cooling systems from mitigating the consequences of any design basis accidents. Therefore, this activity does not increase the consequences of occurrence of an accident previously evaluated in the USAR. Failure or malfunction of the switched jumper will not prevent or affect the ability of safety-related systems or systems important to safety to respond to the accidents described in the USAR. Therefore, consequences of a malfunction of an SSC important to safety previously evaluated in the USAR will not be increased.

Based on a failure modes and effects analysis (FMEA), the potential switch/jumper malfunctions do not create any new DEH responses beyond those already bounded by USAR transient analysis. As such, the possibility of an unanalyzed malfunction of an SSC important to safety or a new type of accident is not created. As described in the USAR, no malfunction of the DEH system can cause a transient sufficient to damage the fuel barrier or exceed the nuclear system pressure limits as required by the safety design basis. Therefore, the modification does not result in the design basis limits for fission product barriers being exceeded or altered. The modification does not result in a departure from a method of evaluation described in the USAR in establishing the design bases or in the safety analysis. Therefore, NRC approval was not required prior to implementing this change.

Engineering Evaluation (EE) 06-038
(Evaluation 2006-0004 Revision 0)

Title: Assessment of Loads Moved Over Irradiated Fuel

Description: License Amendment 222 established revised Technical Specification requirements for secondary containment, control room emergency filter (CREF) system, and associated support systems for fuel movements in accordance with the alternative source term fuel handling accident (FHA). This EE is a companion assessment of certain non-fuel loads that can potentially damage irradiated fuel. This EE establishes the kinetic energies of various non-fuel light loads moved over irradiated fuel during a refueling outage for comparison with the kinetic energy of the dropped fuel bundle in the FHA. The EE modifies the conditions under which the USAR-described design function for secondary containment is to be in place during the movement of non-fuel loads that can damage irradiated fuel.

Secondary containment will not be required for non-fuel load movements whose kinetic energies are bounded by that of the dropped fuel bundle and its handling equipment, provided a 24-hour decay period has elapsed. A similar change to the CREF system design function is created for the CREF system, except that a seven-day shutdown period must elapse before CREF system functionality is no longer required, consistent with the assumptions of the FHA. This change has no effect on the licensing basis with regard to the treatment of heavy loads over or near irradiated fuel. The associated USAR change characterizes the secondary containment and CREF system design function for mitigation of the consequences of load movements that can damage irradiated fuel as a safety design bases.

10 CFR 50.59
Evaluation: A non-fuel light load drop is not an accident described in the USAR. The design basis FHA is unaffected by this proposed change and remains bounding. Therefore, the frequency of occurrence and consequences of previously evaluated accidents are not more than minimally increased. The enhanced design function crediting the CREF system during light load movements does not decrease the reliability of that system to respond, as required, during design basis accidents. Therefore, there is no more than a minimal increase in the likelihood of a malfunction previously evaluated in the USAR and in the consequences of malfunctions previously evaluated.

Since the FHA bounds the consequences of a light load drop and is a more probable event, an accident is not created of a different type than any previously evaluated in the USAR.

The only new possible malfunction is a CREF system on-demand failure. However, the CREF system was designed and licensed as a single train system for which single failure is not assumed for event analysis. Accordingly, a CREF system single failure for light loads analysis is similarly not considered. Therefore, there is not a possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the USAR. Design basis limits for fission product barriers are not affected by this change. There are no affected methodologies described in the USAR used for establishing the design bases or in the safety analyses. Therefore, NRC approval was not required prior to implementing this change.

CED 6019001
(Evaluation 2006-0005 Revision 1)

Title: Install New Screen Wash Strainers

Description: This modification replaced the obsolete strainers on the circulating water (CW) screen wash and sparger systems with two manual and two automatic strainers (one automatic and one manual per system). The replacement strainers fit the flange-to-flange dimensions of the previous strainers and have a finer filter mesh size. The modification also involved installation of additional equipment and equipment anchors, piping, valves, pipe supports, and control cabinets.

10 CFR 50.59
Evaluation: Changing one CW sparger strainer to an automatic backwash strainer will not change the operational characteristics of the CW sparger system. The loss of the CW sparger system due to the failure of the automatic backwash strainer will be no different than the loss due to a manual strainer failure. Failure of the automatic backwash strainer will not result in the loss of CW sparging capabilities due to the redundant manual strainer in the system. New failure modes for the CW sparger automatic backwash strainer are identical to those for the existing CW screen wash automatic strainer and remain bounded by the loss of the CW sparger system. Therefore, a new type of accident or malfunction is not being created.

The CW sparger system does not interface with an SSC important to safety that plays a direct role in mitigating the radiological consequences of an accident. The CW sparger system does not directly interface with nor does it impact the fuel cladding temperature, the reactor coolant system boundary, or containment. The design function of the service water (SW) system (ultimate heat sink) is not adversely impacted and remains capable of performing its design function. No new or existing analysis is required to be generated or changed as a result of this installation. Therefore, NRC approval was not required prior to implementing this change.

CED 6018381
(Evaluation 2007-0001 Revision 0)

Title: Turbine Generator Voltage Regulator Replacement

Description: This modification replaced the main generator voltage regulator with a modern digital voltage regulator. The new voltage regulator is of the same fit, form, and function as the old system and is supplied with dual channel redundancy and diagnostics to provide reliable operation. Equipment inside the voltage regulator cubicles were removed and replaced with the digital control system.

10 CFR 50.59
Evaluation: The FMEA for the replacement turbine generator digital voltage regulator and the turbine generator power system stabilizer demonstrates that no new accidents, accident initiators, or modes of operation are created; the frequency of occurrence of an accident or likelihood of an equipment malfunction is not increased; and that the consequences associated with the failure of the replacement digital voltage regulator and the power system stabilizer are bounded. Additionally, the turbine generator voltage regulator and power system stabilizer are not credited with mitigating the consequences of an accident. Therefore, the frequency of occurrence and consequence of an accident previously evaluated is not increased and the likelihood or consequence of a malfunction of an SSC important to safety is not increased.

The FMEA for the replacement turbine generator digital voltage regulator and the turbine generator power system stabilizer also demonstrates that no new accidents of a different type, or malfunctions of an SSC important to safety with a different result are created, as no new modes of operation, accident initiators, accident consequences, or results of equipment failures are created. The replacement digital voltage regulator and power system stabilizer does not interface or affect reactor coolant or fission product barriers, or change design criteria used in the determination of fission product barrier integrity. Therefore, the design basis limits for fission product barriers are not affected.

The replacement digital voltage regulator and power system stabilizer are not used as design inputs or credited in the methods of evaluation used to establish the safety analysis. Therefore, the installation of the replacement turbine generator digital voltage regulator and the turbine generator power system stabilizer does not result in a departure from a method of evaluation used in establishing the design basis or in the safety analysis. The replacement digital voltage regulator and power system stabilizer does not impact any of the limiting conditions discussed in the Technical Specifications. Therefore, NRC approval was not required prior to implementing this change.

CED 6018240
(Evaluation 2007-0002 Revision 0)

Title: Removal of Single Failure Risk OG-SOV-SSV80 A and B

Description: This modification replaced the off gas (OG) radiation monitor sample return solenoid operated valves (SOV), OG-SOV-SSV80 A and B, with manual valves. This modification removed the single failure risk with one fuse blowing causing both valves to go closed. The new valves meet the design requirements for their locations and will allow adequate sample flow to go through the steam jet air ejector (SJAE) radiation monitoring system.

10 CFR 50.59
Evaluation: This modification replaced the OG radiation monitor sample return SOVs with manual valves. The previously installed SOVs were normally maintained in the open position and were not required to close for any USAR design function. Therefore, removing the SOVs' automatic closure capability and replacing them with normally open manual valves would continue to meet the requirement to have a flow path for the SJAE radiation monitors. The failure mechanism of the signal fusing has been eliminated by this change. No new failure mechanisms impacting USAR design function have been added. From the human system interface perspective, the interaction of an Operator with the manual switch controlling the SOVs is no different with the hand wheel valves. The new configuration is effectively a piece of pipe from an operational perspective. The change in replacing the SOVs with manual valves only improves the reliability of the valves and the system and there are no adverse impacts.

The change did not result in an increase in the frequency of occurrence of an accident previously evaluated in the USAR, nor did it result in an increase in the likelihood of occurrence of a malfunction of the air ejector off-gas radiation monitoring system. Having more reliable valves that remain open only increases the reliability of the air ejector off-gas radiation monitoring system to detect abnormal releases of radioactive materials. Therefore, the consequences of an accident previously evaluated in the USAR were not impacted.

There is no possibility for a malfunction of any SSC associated with the system with a different result than any previously evaluated in the USAR. No accident of a different type nor a failure with a different result have been introduced. The change has no impact on any design basis limit for a fission product barrier, nor does it involve any change in methodology. Therefore, NRC approval was not required prior to implementing this change.

CED 6025820
(Evaluation 2007-0004 Revision 0)

Title: Service Water Booster Pump Interlock Removal

Description: This modification installed an electrical jumper in the "breaker close" logic for each of the four residual heat removal (RHR) service water booster pumps (SWBP). These jumpers will allow the RHR SWBPs to be manually started regardless of the state of the interlock contacts from any of the SW pumps. The modification will improve equipment reliability of the RHR SWBPs by eliminating the potential for a SWBP failure to start due to the mis-operation of an interlock from a SW pump.

10 CFR 50.59
Evaluation: The interlock removal is a reduction in defense in depth assuring adequate net positive suction head (NPSH) for the RHR SWBPs. Existing instrumentation, alarms, and procedural requirements assure proper NPSH prior to starting an RHR SWBP. The installation of the jumpers does not introduce new failure mode effects beyond those previously evaluated in the USAR. The activity does not result in an increase in the frequency of occurrence of an accident previously evaluated in the USAR. The new electrical jumpers increase the reliability of the RHR SWBPs to start when required and will not initiate any new malfunctions. As such, there is no increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR. The new electrical jumpers will increase the reliability of RHR SWBPs to mitigate an accident by eliminating an unnecessary interlock. Therefore, this activity does not increase the consequences of occurrence of an accident previously evaluated in the USAR.

A single failure of a SW pump breaker position switch contact or the electrical jumper will not prevent the start of an RHR SWBP. The consequences of a failure of the RHR SW booster system remain unchanged. Therefore, the change will not increase the consequences of a malfunction of an SSC important to safety previously evaluated in the USAR. The RHR SW booster system breakers and pumps are not accident initiators. Therefore, the activity does not create the possibility of an accident of a different type than previously evaluated in the USAR.

Based on a FMEA, the potential electrical jumper malfunction does not create any new RHR SW booster system malfunction results beyond those already bounded by the failure of one RHR SWBP. Therefore, the possibility of an unanalyzed malfunction of an SSC important to safety is not created. The modification does not result in the design basis limits for fission product barriers being exceeded or altered. The modification does not result in departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses. Therefore, NRC approval was not required prior to implementing this change.

Temporary Configuration Change (TCC) 4608583
(Evaluation 2008-0001 Revision 0)

Title: Installation of Temporary Control Air Supply to LCV 67B

Description: This modification installed a temporary air supply to bypass the positioner for the level control valve (LCV) to the B3 feedwater heater. The temporary control air supply will allow manual control to this valve so that level to the B3 feedwater heater can be appropriately maintained.

10 CFR 50.59
Evaluation: Changing the operation of the LCV from automatic to manual does not introduce the possibility of a change in the frequency of an accident as failure of the air system would result in no new failure mode for the valve. On a loss of air, the valve would still go open. Failure of the newly installed pressure regulator would be considered equivalent to a failure of the previous positioner as both perform the same functions. Replacing the positioner with a manual controlled pressure regulator does not increase the likelihood of occurrence of a malfunction. The equipment in the system would function as designed and malfunction would not be more likely as a result of this change.

The manual operation of the LCV does not introduce the possibility of a change in the consequences of an accident or malfunction because failure of the valve would not change any of the accident conditions or result in an increase of dose. This temporary modification does not introduce new failure modes. As such, the possibility of a new accident is not created nor is the possibility for a malfunction of an SSC important to safety with a different result. Feedwater heater failure could impact a fission product barrier as described in the USAR. Colder water would cause a transient that could affect the barrier. However, the change made by this temporary modification does not exceed or alter the design basis limit as described in the USAR.

None of the SSCs affected by the modification are credited in the safety analysis nor do they support or impact an SSC function credited in the safety analysis. Therefore, the change does not result in a departure from a method of evaluation. Therefore, NRC approval was not required prior to implementing this change.

TCCs 4614562 and 4636337
Evaluations (2008-0005 Revisions 0 and 1)

Title: Defeat OWC Hi H_2 Flow Shutdown

Description: These TCCs implemented actions to defeat the high hydrogen flow automatic shutdown of the optimum water chemistry (OWC) hydrogen injection system. The temporary change is to prevent unnecessary shutdowns of hydrogen injection due to problems with moisture in the process stream affecting the flow element, causing spurious high flow indication. Other manual and automatic shutdowns for the system remain.

10 CFR 50.59
Evaluation: The OWC system does not have the ability to change the frequency of occurrence of any accident previously evaluated in the USAR since it does not have a safety function or design basis. The OWC hydrogen injection system is isolated from the other systems with which it interfaces by isolation devices so that it cannot affect a safety system. As such, there is no credible way that OWC can change the likelihood of any malfunction of any SSC important to safety. The defeat of the high hydrogen flow shutdown does not change this relationship.

The OWC system and hydrogen injection do not create a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that results in a fuel cladding failure. Therefore, the possibility of an accident of a different type from any previously evaluated in the USAR is not created. OWC cannot affect a plant operating condition or plant operation with respect to nuclear safety. The transient analysis in the USAR will not be affected. Hydrogen injection cannot create a new safety-related equipment failure mode, the possibility of a new limiting transient, or new sequence of events that can result in a radiological release above current operating or regulatory limits.

No changes to any of the analysis or methods of analysis used in establishing the design bases or in the safety analyses are required for defeating the high hydrogen flow shutdown. Therefore, NRC approval was not required prior to implementing this change.

Revision 1 to Evaluation 2008-0005 was required to implement this activity again under a new TCC.

EE 08-014
(Evaluation 2008-0007 Revision 0)

Title: Control Rod Drop onto Spent Fuel Storage Pool Analysis

Description: In September 2007, permission was received from the NRC to expand the storage space in the spent fuel pool. The USAR describes the results of analysis performed to determine the impact of a control rod blade dropped in the spent fuel pool. One of the assumptions is that there is a clear path to transfer control rod blades to the pool without moving an irradiated blade directly above a fuel bundle. The addition of the new storage space in the spent fuel pool has precluded this assumption. This EE was performed to review vendor analysis of the consequences of a control rod blade drop onto a fuel bundle while being transferred to or from the spent fuel pool.

10 CFR 50.59
Evaluation: The accident previously described in the USAR which would be similar to this change is the FHA. The only components which would affect the frequency or likelihood of occurrence of this accident are those on the refueling platform. This change does not alter these components in any way. As such, there is no change to the frequency or likelihood of occurrence of the FHA. As this change involves a blade drop over the spent fuel pool, there is no change in consequence for the FHA as currently evaluated in the USAR. The consequences of the FHA bound this change. The only components that could malfunction are those on the refueling platform. Therefore, there is no change to the consequences of their malfunction as reflected in the FHA since these components were not altered.

While a control rod blade drop onto fuel is not explicitly evaluated in the USAR, the event is similar in assumption and type to the FHA. The item dropped and the height of the drop are the main differences, but the plant configuration (primary containment open with release to secondary containment) and the evaluation of the energy imparted to the fuel cladding being performed in the same manner makes this event of the same type as the FHA. A refueling platform failure is a type of malfunction evaluated in the FHA. As the FHA is previously evaluated in the USAR, there is no possibility for a malfunction of the SSC to occur with a different result.

None of the design basis limits for fission product barriers are affected by this change and the method of evaluation for the design basis and the safety analyses is not altered. Therefore, NRC approval was not required prior to implementing this change.

ATTACHMENT 2

OTHER CHANGES

Special Procedure (SP) 07-001
(Evaluation 2008-0002 Revision 0)

Title: Main Generator Ventilation and Startup Testing and Tuning of Main Generator Voltage Regulator

Description: This temporary SP performed operational testing and performance tuning of the rewound main generator ventilation system and new generator voltage regulator installed via Change Evaluation Documents (CED) 6018380 and 6018381. During plant startup from refueling outage 24, the SP performed testing of the main generator ventilation while the generator was filled with air versus hydrogen prior to placing the generator in service. This test was necessary to assure proper ventilation through the generator. This SP also covered testing and fine tuning of the voltage regulator control system.

10 CFR 50.59 Evaluation: The SP does not increase the frequency of occurrence of previously analyzed accidents as its failure is not an initiator of any design basis accident and no new failure modes are created to increase the frequency of abnormal transients involving the generator ventilation or generator excitation systems. The SP does not increase the likelihood of occurrence of a malfunction previously evaluated as no new failure modes are introduced. The SP does not increase the consequences of any previously evaluated accidents or malfunctions as the generator ventilation or generator excitation systems are not credited mitigation systems.

The SP does not affect any design basis limits for fission product barriers and does not involve any new methods of evaluation used in establishing the design bases or in the safety analyses. Therefore, Nuclear Regulatory Commission (NRC) approval was not required prior to implementing this change.

SP 06-001
(Evaluation 2008-0003 Revision 0)

Title: Operational Testing Reactor Vessel Level and Reactor Feedpump Turbines A and B Control Systems

Description: This temporary SP performed operational testing and performance tuning of the new reactor vessel level control (RVLC) system and reactor feedpump turbine (RFPT) speed control system installed under CEDs 6010820 and 6016542. Much of the new RVLC system and RFPT control system were tested under the cognizance of installation for the CEDs by inputting electronic signals to simulate an operating system. Actual system dynamic testing and tuning was required and performed per this SP. During plant startup, the SP monitored various system parameters during controlled level and speed setpoint changes to determine system performance characteristics. In addition, RFPT vibration instrumentation was monitored to determine the critical speed ranges. Test equipment was temporarily connected to facilitate speed sensor monitoring and startup valve tuning. Appropriate control parameters within the new RVLC system and RFPT control system were adjusted to fine tune system performance. New reactor feedpump and turbine vibration equipment installed via CED 6016542 was monitored during the plant startup from refueling outage 24 and new alarm setpoints were determined and programmed.

10 CFR 50.59 Evaluation: The SP does not increase the frequency of occurrence of previously analyzed accidents as its failure is not an initiator of any design basis accident and no new failure modes are created to increase the frequency of abnormal transients involving the feedwater system. The SP does not increase the likelihood of occurrence of a malfunction previously evaluated as no new failure modes are introduced. The SP does not increase the consequences of any previously evaluated accidents or malfunctions as the feedwater system is not a credited mitigation system.

The SP does not affect any design basis limits for fission product barriers and does not involve any new methods of evaluation used in establishing the design bases or in the safety analyses. Therefore, NRC approval was not required prior to implementing this change.

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©

Correspondence Number: NLS2008094

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		