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NLS2004117
October 14, 2004

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

Subject: 10CFR50.59(d)(2) Summary Report
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

The purpose of this letter is to provide the summary report of facility/procedure changes, tests, and experiments that have been completed (Attachments 1, 2, and 3), in accordance with the requirements of 10CFR50.59(d)(2). This report covers the time period from August 1, 2002, to July 31, 2004. All of the 10CFR50.59 evaluations reported herein were performed using the current 10CFR50.59 Rule.

In accordance with 10CFR50.4, 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 my office at (402) 825-2774.

Sincerely,

Paul V. Fleming
Licensing Manager

/wrv
Attachments

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

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

Senior Resident Inspector, w/attachments
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ATTACHMENT 1

FACILITY CHANGES

CED 6007522 and Change Notice 5
(Evaluation 2001-0039 and 2001-0039 Rev. 1)

TITLE: Reactor Feedwater Control System Upgrade Phase 1

DESCRIPTION: CED 6007522 replaces the existing analog Reactor Feedwater (FW) Control System with digital equipment. This modification is being implemented to improve the availability and reliability of the Reactor Feedwater Pump Turbine (RFPT) speed control and startup valves level control systems. The major elements of this modification are:

- Replacement of existing RFPT speed control system electronics, including Control Room panel hardware.
- Replacement of existing analog control components in the Startup Level Control loops for reactor feed startup valves with digital equipment (with the exception of the level transmitters and power supplies).
- Enhanced system monitoring capability, including an Operator Information Touchscreen.
- Addition of power trip relays that transfer the RFPT speed control system to manual demand mode on loss of power.

The current RFPT speed control system control functions and performance requirements are unchanged with the exception of: a) elimination of FW flow limit control function, b) removing track and hold control circuit, c) removing lock-out relay from the loss FW demand signal and loss of controlling level loop circuit, and d) the addition of Isochronous Speed Correction.

Change Notice 5 resulted in administrative changes to Evaluation 2001-0039 from the identification of additional affected procedures.

10CFR50.59
EVALUATION: The main control functions of the RFPT speed control system remain as described in the Updated Safety Analysis Report (USAR). The only USAR-described events for which RFPT speed control and startup level control system failures apply are: a) Loss of Feedwater Flow, b) Feedwater Controller Failure Maximum Demand, and c) ATWS- Loss of Normal Feedwater. No new failure modes are introduced by this upgrade, and there is enhanced reliability. Radio Frequency Interference and Electro-Magnetic Interference (RFI/EMI) was considered in the design of this modification per TR-102323, "Guidelines for Electromagnetic Interference Testing of Power Plant Equipment" and TR-102400, "Handbook for Electromagnetic Compatibility of Digital Equipment in Power Plants." Therefore, there is neither an increase in the frequency of occurrence of an accident nor an increase in the likelihood of a malfunction previously evaluated in the USAR. The RFPT speed control system performance requirements and failure mode effects of the new system are unchanged, and the non-safety-related system is not credited for mitigating the consequences of USAR-described accidents. Therefore, there is no increase in consequences of previously evaluated accidents or of malfunctions described in the USAR. Additionally, as part of the RFI/EMI considerations, appropriate grounding and shielding were designed into the signal wiring and components. Therefore, the possibility of an accident of a different type is not introduced. The new equipment will not initiate any new malfunctions or failures with a different result as previous evaluated in the USAR. This modification does not result in a design basis limit for a fission product barrier being exceeded or altered, nor result in a departure from a USAR-described methodology. Finally, this modification does not require a change to the Technical Specification

or a change to the Operating License. Therefore, this change may be made without prior Nuclear Regulatory Commission review and approval.

CED 1999-0072 Change Notices 32, 45, and 46
(Evaluation 2002-0005 Revisions 0 through 3)

TITLE: Optimum Water Chemistry (OWC) Modification

DESCRIPTION: CED 1999-0072 was reported in the last 10CFR50.59 Summary Report. Change Notice 32 performs the following changes to the OWC Gas Generation and Injection systems to improve the reliability of the system and to perform testing of the system without injecting any gases into the plant:

- Revise the setpoint for H_2 in O_2 analyzer.
- Modify the P700 PLC logic to eliminate the auto-restart of hydrogen compressors.
- Install a hydrogen flow indication subsystem.
- Install a hydrogen vent line in the Turbine Building north wall.
- Document the Environmental Qualification (EQ) impacts due to elevated background radiation levels associated with hydrogen injection.
- Add an evaluation and implement a modification for potential high Main Steam Line Radiation Monitor levels associated with a hydrogen flow transient.

Testing has been performed to verify the proper operation of the system components without injecting any gas into the plant. The final test phase is to review plant response to the actual injection of hydrogen and oxygen. Revisions to temporary Special Procedures and temporary revisions to existing Operations procedures have been made to support the gas injection testing into the plant.

Change Notice 45 updates Special Procedures to incorporate changes to the OWC Gas Generation and Gas Injection system that are based on resolutions to system operating problems uncovered during testing. The changes provide: a) a permanent Turbine Building hydrogen vent path, b) a filtered Instrument Air supply to OWC-RACK-P200, and a higher Main Steamline Radiation Monitor (MSLRM) administrative limit to allow continued OWC testing to minimize actuation of the MSLRM HIGH alarm setpoint.

Change Notice 46 alters the P500PLC logic (control logic for the OWC Gas Injection system) to remove the LOW OXYGEN PRESSURE shutdown signal to allow troubleshooting and monitoring of the oxygen skid. The LOW OXYGEN PRESSURE shutdown stops hydrogen injection as well as oxygen injection. To minimize the loss of hydrogen injection and minimize changes to plant chemistry due to problems with the oxygen compressor skid, this shutdown needs to be temporarily eliminated until a permanent solution is supplied by General Electric.

10CFR50.59
EVALUATION: Evaluation 2002-0005 supplements previously reported Unreviewed Safety Question Evaluation 2000-0007 to reflect the new 10CFR50.59 Rule, as well as to reflect the above changes to CED 1999-0072. This summary combines the two Evaluations to provide an integrated discussion of CED 1999-0072 with respect to 10CFR50.59.

OWC is a proven technology for the prevention of Inter-Granular Stress Cracking Corrosion (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 oxygen 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 are 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 RCS will remain sufficiently high to prevent flow accelerated corrosion; therefore, the probability of a Main Steam Line Break Accident (MSLBA) 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 was determined to be 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 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, oxygen, and plant air (only 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. Low oxygen pressure results in Control Room annunciation. 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 oxygen 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.

EE02-026
(Evaluation 2002-0014)

TITLE: Implementation of Drywell Average Temperature

DESCRIPTION: The activity consists of implementing Engineering Evaluation EE02-026 which changes the formula that is used to compute drywell average air temperature during normal operation under certain conditions. The conditions specifically require Mode = 1 (Run), all four (4) drywell fan coil units operable and in operation, and specified temperature elements to be operable. If any condition is not met, then the formula reverts back to the pre-existing method. The new formula is based on a dynamic model which gives more accurate results and a better documented technical basis than the old formula that is based on a static model.

10CFR50.59
EVALUATION: The proposed activity was reviewed and found to be neither a precursor nor initiator. Therefore, it has no effect on frequency of occurrence of an accident or on the likelihood of any malfunction of any Structure, System or Component (SSC). The activity does not introduce new failure modes or malfunctions. Therefore, there is no change in the consequences of any accident or malfunction of an SSC. It also does not introduce a new accident or a malfunction with a different result. The activity does not change any design input value. Therefore, it does not affect any Design Basis Limit for Fission Product Barriers. The activity also does not result in a departure from a method of evaluation used in establishing the design basis or in the safety analyses. It does not require a Technical Specification change or License Amendment, and therefore may proceed without prior Nuclear Regulatory Commission approval.

CED 6005412 Change Notices 11 and 12
(Evaluation 2003-0002, Revisions 0 and 1)

TITLE: Addition of Alarm Time Delays to the Service Water (SW) Radiation Monitor System

DESCRIPTION: The SW radiation monitoring system was replaced under CED 6005412. The system is much more sensitive to background radiation. Therefore, Change Notices 11 and 12 add time delays to the High Radiation alarm and the Inoperable (Inop) alarm to prevent nuisance alarms from becoming an operator distraction. The time delays will eliminate short duration alarms caused by changes in radiation background due to operational processes affecting, or work activities occurring in the vicinity of, the radiation monitors.

10CFR50.59
EVALUATION: The alarm time delay is not an initiator of any accident (or new malfunction), and its addition does not introduce any new failure modes. Additionally, the activity does not create a new mode of plant operation, or create any new equipment interfaces that could affect the likelihood of an accident (or equipment malfunction) or its consequences. Therefore, the activity does not increase the frequency of occurrence of an accident or the likelihood of occurrence of a malfunction previously evaluated in the Updated Safety Analysis Report (USAR). Although there is a potential for an increase in the dose consequence associated with the High Radiation alarm time delay, it is bounded by the existing effluents license basis described in the Cooper Offsite Dose Assessment Manual. The Inop alarm time delay does not affect the radiation monitor's ability to immediately detect a high liquid effluent activity and alarm, and does not affect the dose consequence results evaluated as a result of the radiation monitor's alarm time delay. The alarm time delay does not affect any other type of accident or accident mitigation feature described in the USAR since it is not an initiator of an accident and has no interface with any other type of accident mitigating features. Therefore, the activity does not result in a more than minimal increase in the consequences of an accident previously evaluated in the USAR. Since the alarm time delay is not an initiator of any new or different type of accident and the alarm time delay is not an initiator of

any new malfunction, the activity does not create the possibility for an accident of a different type than any previously evaluated in the USAR, and also does not create a possibility for a malfunction of a Structure, System or Component important to safety with a different result than that previously evaluated in the USAR. The alarm time delay does not result in any design basis limit for a fission product barrier being exceeded or altered, and does not result in any changes to USAR-described methodologies.

TCC 4301609
(Evaluation 2003-0006)

TITLE: Lock Open Devices for Mechanical Overspeed Butterfly Valve on Diesel Generator (DG) 1

DESCRIPTION: Temporary Configuration Change (TCC) 4301609 was implemented to install a gag (locking) device in order to maintain the DG 1 right bank air inlet butterfly valve in the open position. This TCC was required as a result of damage to the valve control cable. Procurement of a replacement cable will take several weeks. The right bank air inlet butterfly valve was gagged open to support DG 1 testing and operability until the replacement cable can be installed.

10CFR50.59
EVALUATION: Since the DGs are not an initiator of any of the abnormal operating transients or postulated accidents described in the Updated Safety Analysis Report (USAR), this temporary configuration change does not increase the possibility of a change in the frequency of an accident previously evaluated in the USAR. DG 1 retains the safety shutdown features and emergency operation/functions as specified in the USAR and does not more than minimally increase the possibility in the likelihood of a malfunction of an SSC important to safety previously evaluated in the USAR. This modification will not adversely affect the ability of DG 1 to provide emergency power to Engineered Safety Feature systems used for accident mitigation and therefore, does not increase the consequences of occurrence of an accident previously evaluated in the USAR. This TCC will not change the consequences of a failure of a DG on any other safety system as stated in the USAR and does not increase the consequences of a malfunction of a System, Structure or Component (SSC) important to safety. Changes to the DG cannot create the possibility of an accident of a different type (i.e., the DGs are a mitigation system) and as such there is no increase in the possibility of an accident of a different type than any previously evaluated in the USAR. The USAR assumes that only one onsite DG is available during the entire design basis Loss-of-Coolant Accident. Therefore, the possibility of a malfunction of an SSC important to safety with a different result from any previously evaluated in the USAR will not be created. The change incorporated by this TCC does not affect the accident analysis for the release of radioactive material and does not affect any of the radioactive material. The proposed activity does not result in design basis limit for a fission product barrier as described in the USAR being exceeded or altered. This TCC does not result in a departure from a method of evaluation described in the USAR used in establishing the design basis or in the safety analysis.

EE03-024
(Evaluation 2004-0001)

TITLE: Revised Reactor Building High Energy Line Break (HELB) and Post-LOCA Heatup (PLHU) Temperature and Pressure Profiles for Environmental Qualification (EQ) Purposes

DESCRIPTION: EE03-024 implements new Reactor Building HELB and PLHU analyses resulting in new temperature and pressure profiles for EQ purposes. The new analyses are based on the latest uniform set of design inputs, conservative assumptions, using updated GOTHIC 7.0 computer program, and more accurate computer modeling of the Cooper Nuclear Station plant. Previous analyses used an earlier GOTHIC version and other software and hand calculations. This is only a change to the EQ basis temperature profiles.

10CFR50.59
EVALUATION: This Evaluation only concerns a change in a methodology described in the Updated Safety Analysis Report. The supporting calculations use a later GOTHIC computer code version, or a new code (GOTHIC instead of RELAP) which is a change in an element of methodology for developing temperature and pressure profiles throughout the Reactor Building for EQ purposes. The method is still the same, i.e., by analysis. The results were compared to previous results and found to be conservative or essentially the same.

EE03-050
(Evaluation 2004-0002)

TITLE: Environmental Qualification Total Integrated Radiation Dose Analysis

DESCRIPTION: As part of the Environmental Qualification (EQ) Improvement Project, the existing calculations for the radiation environment analysis were reviewed. A number of discrepancies were noted with the design input values, a mixture of methodologies were applied without a well-documented basis, and the original contractor software was not provided to Cooper Nuclear Station (CNS). Thus, a revised CNS EQ analysis is being implemented under EE03-050. It uses: a) software approved for use at CNS via the CNS Software Quality Assurance Plan and used by other licensees in support of License Amendment Requests to the Nuclear Regulatory Commission (NRC), b) the CNS Loss-of-Coolant Accident (LOCA) analysis and source term methodology reviewed and approved by the NRC in CNS License Amendment 196, c) the widely accepted EQ assumptions established in the Boiling Water Reactor Owners' Group (BWROG) Position Paper for Shielding and Environmental Qualification, d) the widely used and accepted Loevinger beta dose methodologies, and e) more recent normal operation plant radiation data. The results of the EQ normal radiation environment and accident dose analysis implemented by this Engineering Evaluation will be used as input to the EQ equipment Total Integrated Dose (TID) qualification data evaluation and replace, in its entirely, the existing EQ radiation dose analysis.

10CFR50.59
EVALUATION: None of the accident or Structure, System or Component (SSC) malfunction initiators or failure modes assumed in the LOCA analysis are affected by this EE. Thus, mitigation features and assumptions in the LOCA analysis are unaffected. Additionally, the assumptions and methodology of EE03-050 use the existing plant design to conduct the EQ radiation dose analysis and do not involve any change to plant operating procedures, operating modes, system performance or plant configuration. EE03-050 and its calculations do not have any impact on any of the other accidents, equipment malfunctions, or mitigation functions previously evaluated in the Updated Safety Analysis Report since the LOCA is used as the bounding event for the environmental qualification radiation dose analysis.

EE03-050 does not create any new accident initiators, SSC malfunctions, or failure modes since the analyses of EE03-050 assume that a LOCA has already occurred. EE03-050 uses existing plant design to conduct the environmental qualification radiation dose analysis and does not involve any change to plant operating procedures, operating modes, system performance or plant configuration. Thus, no new accident initiators, SSC malfunctions or failure modes are created.

EE03-050, as with the existing EQ accident radiological integrated dose calculations it replaces, continues to use a method of evaluation to evaluate EQ accident and normal environment radiological dose consistent with Regulatory Guide 1.89. The elements of the methodology, however, were revised in part to reflect recent changes in the CNS license/design basis, correct errors in the existing EQ analyses, use up-to-date software previously used by licensees in support of License Amendment Request submittals to the NRC, use widely accepted EQ assumptions established in the BWROG Position Paper for Shielding and Environmental Qualification, use widely accepted Loevinger beta dose evaluation methodologies, and use more recent normal

operation plant radiation data. These changes result in a revised EQ TID analysis that has results that are essentially the same or more conservative than the existing EQ TID analysis.

Therefore, EE03-050 and its associated calculations/recommendations do not require prior NRC approval.

ATTACHMENT 2

PROCEDURE CHANGES

Temporary Procedure SP02-004
(Evaluation 2003-0003)

TITLE: Operational Testing Startup Level Control and Reactor Feed Pump Turbines A and B Control

DESCRIPTION: Temporary Procedure SP02-004 performs operational testing and performance tuning of the new reactor feed pump turbine and startup level control systems installed by CED 6007522 and new startup flow control valves installed by CED 6008140. Although much of the new feedwater control system will be tested under the cognizance of the installing CED by inputting electronic signals to simulate an operating system, actual system dynamic testing is required. During plant startup from RE21, SP02-004 will monitor various system parameters during controlled level setpoint changes to determine system performance characteristics. Test equipment will be temporarily connected to facilitate monitoring. Appropriate control parameters within the new feedwater control system will be adjusted to fine tune system performance. New reactor feed pump and turbine vibration equipment installed via CED 6007522 will also be monitored during the plant startup. New alarm setpoints will be determined and programmed, as necessary. Successful completion of SP02-004 will demonstrate proper system functionality, stability and performance.

10CFR50.59
EVALUATION: Temporary Procedure SP02-004 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. SP02-004 does not increase the likelihood of occurrence of a malfunction previously evaluated as no new failure modes are introduced. SP02-004 does not increase the consequences of any previously evaluated accidents or malfunctions as the feedwater system is not a credited mitigation system. SP02-004 does not affect any design basis limit for a fission product barrier and does not involve any new methods of evaluation used in establishing the design bases or in the safety analyses. In summary, SP02-004 has been evaluated against the eight criteria of 10CFR50.59(c)(2) and determined that prior NRC approval is not required.

Procedure 10.33 (Revision 7)
(Evaluation 2004-0003)

TITLE: Relaxation of Upper Peak Cladding Temperature Limit

DESCRIPTION: The Nuclear Regulatory Commission (NRC) Safety Evaluation approving the SAFER/GESTR-LOCA methodology for use at Cooper Nuclear Station (CNS) placed a restriction of 1600 degrees F on the upper bound peak cladding temperature (PCT) calculation. The upper bound PCT was originally provided to demonstrate that the licensing basis PCT was sufficiently conservative. General Electric has received NRC approval to eliminate this restriction generically for SAFER/GESTR-LOCA. Procedure 10.33 is being revised to eliminate the 1600 degree F PCT limit.

10CFR50.59
EVALUATION: This procedure change only involves a change in methodology. CNS has a plant specific NRC Safety Evaluation approving the use of the SAFER/GESTR-LOCA methodology. The NRC has generically approved elimination of the PCT restriction for SAFER/GESTR-LOCA. Therefore, it is concluded that there is an NRC Safety Evaluation that can be applied to CNS for the intended application, and that this methodology change may be made without prior NRC approval.

ATTACHMENT 3

OTHER CHANGES[1]

OLCR 2001-028
EE02-033
(Evaluation 2002-0010)

TITLE: Licensing Basis Document and Procedure Changes to Implement Proposed Operating License Change Request to Increase Ultimate Heat Sink and Reactor Equipment Cooling System Temperature

DESCRIPTION: License Amendment 193 approved changes to the Technical Specification (TS) limits for river supply temperature from 90°F to 95°F, and Reactor Equipment Cooling (REC) supply temperature from 95°F to 100°F. This requires concurrent changes to the TS Bases. At the time this TS Bases change was implemented, Station procedures required performance of a 10CFR50.59 Evaluation regardless of whether there was an associated License Amendment. Engineering Evaluation EE02-033 was generated to identify the affected documentation, and to insure that the intent of Amendment 193 is properly incorporated into the Cooper Nuclear Station (CNS) licensing and design basis documents. Several elements of EE02-033 were not specifically discussed in the License Amendment 193 correspondence, and thus, are being implemented under the auspices of 10CFR50.59. These elements are:

- A reduction in time required to initiate suppression pool cooling during an Appendix R Shutdown From Outside the Control Room event from 3 hours to 1.5 hours.
- Procedural changes in maximum Turbine Equipment Cooling (TEC) operating temperature from 95°F to 100°F.
- Procedural changes to add guidance to maintain Service Water flow to the REC heat exchangers or isolate the non-critical REC loads if temperature indication is lost and indications of a primary coolant leak are present.
- The temperatures and pressures of various piping systems are increased as a result of the river water and REC temperature increases.

10CFR50.59
EVALUATION: The TS Bases changes associated with License Amendment 193 were provided to the Nuclear Regulatory Commission (NRC) for information with the License Amendment Request. Thus NRC approval of the TS change bounds the changes made to the TS Bases.

The following points relate to the additional changes:

- The reduction in time required to implement suppression pool cooling during an Appendix R fire has no effect on the frequency of occurrence of that event. The actual time necessary to implement suppression pool cooling is approximately 1 hour. Thus, the revised time remains conservative.
- Although the TEC maximum allowable temperature is increased, this does not result in any equipment limit or alarm setpoint increases. Appropriate actions remain in place to maintain these limits. The short durations of high temperatures and the slow predictable changes in river temperature do not create excessive operational burdens in maintaining these equipment limits.

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1. This attachment includes the following types of change activities: Technical Requirements Manual Change Requests (TRMCRs) and Operating License Change Requests (OLCRs).

- The procedural changes associated with adding guidance to maintain specified Service Water flow to the REC heat exchangers are considered procedural enhancements. They offer additional guidance to operators to maintain analyzed conditions in the event of failed temperature indication. These new steps are only entered if a Group 2 isolation signal due to a potential Loss-of-Coolant Accident.
- The increase calculated temperatures and pressures resulting from increased river and REC temperature do not result in exceeding code allowable stress values for these systems.

In summary, these changes do not create more than a minimal increase in the frequency of occurrence of an accident and do not pose a potential for increase in the likelihood of occurrence of a malfunction of equipment important to safety. Similarly, consequences of previously evaluated accidents and malfunctions will not increase, nor will the possibility of an accident of a different type or a malfunction with a different result be introduced. This change presents no challenges to design basis limits for fission product barriers and does not result in changes to methodologies described in the Updated Safety Analysis Report.

OLCR 2002-017
(Evaluation 2002-0011)

TITLE: Technical Specifications Bases Change to Implement Proposed Operating License Change Request to Remove Details of Method of Recording Reactor Equipment Cooling Surge Tank Level

DESCRIPTION: License Amendment 194 approved a revision to Technical Specification (TS) Surveillance Requirement 3.7.3.1. This License Amendment removed details of the method of verifying Reactor Equipment Cooling (REC) surge tank level, i.e., the reference to the surge tank gauge glass. The Limiting Condition for Operation section of TS Bases B 3.7.3 contains mention of the surge tank gauge glass, and this section is revised to delete specific reference to the gauge glass. At the time this TS Bases change was implemented, Station procedures required performance of a 10CFR50.59 Evaluation regardless of whether there was an associated License Amendment.

10CFR50.59
EVALUATION: This change rewords the specific operability statement in the TS Bases section that describes REC system operability. Neither the current wording nor the proposed wording will affect the performance of the minimum equipment credited in the mitigation of any analyzed event. This change will not make any changes to any analysis supporting current system design and/or operation. The Updated Safety Analysis Report defines the maximum allowed leakage. The minimum criteria for REC leakage is not changed nor is frequency of checking that the leakage criteria are met changed. There is no change in actual REC system analysis, design, or operation. This TS Bases change is being performed as part of License Amendment 194 implementation. Thus, no additional level of prior Nuclear Regulatory Commission review and approval is required.

OLCR 2002-011
(Evaluation 2003-0005)

TITLE: Technical Specification Bases Change to Implement Proposed Operating License Change Request to Adopt TSTF-358

DESCRIPTION: The Technical Specification (TS) Bases for surveillance Requirement (SR) 3.0.3 are being revised as part of implementing License Amendment 197, issued on March 6, 2003. Amendment 197 issued a revised TS SR 3.0.3 that allowed an increase in the period of time for performing a Surveillance Requirement (SR) that was discovered to have not been performed when required. The TS Bases revision implemented by this 10CFR50.59 Evaluation is a required part of the

revision of SR 3.0.3 adopted by the Nuclear Regulatory Commission (NRC) as part of the Consolidated Line Item Improvement Process (CLIIP). At the time this TS Bases change was being implemented, Cooper Nuclear Station (CNS) procedures required completion of a 10CFR50.59 Evaluation regardless of whether the change was associated with a License Amendment.

10CFR50.59
EVALUATION: As part of the CLIIP, the NRC staff prepares a model application that licensees must follow in preparing the license amendment request for their plant. The model application included proposed TS Bases changes, and required that licensees commit to implement those TS Bases as part of implementing the amendment when issued. The CNS submittal requesting the SR 3.0.3 change (NLS2002098, dated 9/26/02) contained the TS Bases that were part of the model application. This 10CFR50.59 Evaluation implements the TS Bases revisions that were included in the CNS license amendment request and approved by the NRC as issued by Amendment 197. The TS Bases changes evaluated by this Evaluation were reviewed by the NRC as part of their review of the CNS application for changing SR 3.0.3. Thus, additional prior NRC approval is not required under the provisions of 10CFR50.59.

TRMCR 2002-002
(Evaluation 2003-0013)

TITLE: Technical Requirements Manual (TRM) Changes to Sections TSR3.7.1, and B3.7.2.

DESCRIPTION: TRM Surveillance Requirement TSR 3.7.1 and TRM Bases B3.7.2 are being changed to refer to 10CFR50.55a rather than 10CFR50.55a(g)(6)(i). There are six paragraphs, not just the one presently being referenced, in which a licensee may seek relief or deviation from Section XI Code requirements. Additionally, a paragraph is being deleted from B3.7.2 which discusses deviating from procedures prescribed in Section XI, including reporting such deviations to the Nuclear Regulatory Commission (NRC). Section XI does not prescribe procedures, but stipulates requirements. Furthermore, deviation from the Code is not reported to the NRC. Rather, the NRC must grant approval under the appropriate paragraph of 10CFR50.55a for any deviation from a Code requirement.

10CFR50.59
EVALUATION: Station procedures require a 10CFR50.59 Evaluation be performed for non-editorial changes to the TRM. The proposed changes to the TRM are essentially administrative changes that do not alter the requirements of 10CFR50.55a for seeking NRC approval for relief or deviation from ASME Section XI. Therefore, it has been concluded that these TRM changes can be made without prior NRC approval.

Correspondence Number: NLS2004117

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	COMMITTED DATE OR OUTAGE
None	