July 14, 2004

Mr. Randall K. Edington
Vice President-Nuclear and CNO
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT ON ONE-TIME EXTENSION OF SURVEILLANCE REQUIREMENTS (TAC NO. MC1914)

Dear Mr. Edington:

The Commission has issued the enclosed Amendment No. 205 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 30, 2004, as supplemented by letter dated June 17, 2004.

The amendment would revise a limited number of TS Surveillance Requirements (SRs) to allow a one-time extension. The core will not be fully utilized by the originally planned October 2004 outage, which has been rescheduled for January 2005. The SRs identified in the submittal would come due during the extended current cycle operation or before an opportunity to perform them during the January 2005 outage.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Michelle C. Honcharik, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 205 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

Mr. Randall K. Edington
Vice President-Nuclear and CNO
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

July 14, 2004

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT ON ONE-TIME EXTENSION OF SURVEILLANCE REQUIREMENTS (TAC NO. MC1914)

Dear Mr. Edington:

The Commission has issued the enclosed Amendment No. 205 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 30, 2004, as supplemented by letter dated June 17, 2004.

The amendment would revise a limited number of TS Surveillance Requirements (SRs) to allow a one-time extension. The core will not be fully utilized by the originally planned October 2004 outage, which has been rescheduled for January 2005. The SRs identified in the submittal would come due during the extended current cycle operation or before an opportunity to perform them during the January 2005 outage.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,
/RA/
Michelle C. Honcharik, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 205 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:
PUBLIC
PDIV-1 r/f
GHill (2)
RidsNrrDlpmLpdiv (HBerkow)
RidsNrrDlpmLpdiv1 (RGramm)
RidsNrrPMMHoncharik
RidsNrrLAMDBaxley
RidsNrrDipmIrob (TBoyce)
RidsAcrsAcnwMailCenter
RidsOgcRp
RidsRgn4MailCenter (KKennedy)
BGrimmel
FAkstulewicz
HWalker
RDennig
SMazumdar
RJenkins
SRhow
EMarinos
RidsNrrDlpmDpr(SWall)

Package No.: ML041980576
TS Pages: ML041970592 **Nrr-100**
ACCESSION NO: ML041960078 **Nrr-058** *No substantive changes made

OFFICE	PDIV-1/PM	PDIV-1/LA	OGC	PDIV-1/SC
NAME	MHoncharik	DBaxley	RHoefling	RGramm
DATE	7/2/04	7/2/04	7/12/04	7/13/04
OFFICE	EEIB-A/SC	EEIB-B/SC	SRXB/SC	SPSB/SC
NAME	EMarinos	RJenkins	FAkstulewicz	RDennig
DATE	4/19/04*	5/7/04*	4/12/04*	4/26/04*

OFFICIAL RECORD COPY

DOCUMENT NAME: C:\ORPCheckout\FileNET\ML041960078.wpd

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 205
License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nebraska Public Power District (the licensee) dated January 30, 2004, as supplemented by letter dated June 17, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 205, are hereby incorporated in the license. The Nebraska Public Power District shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 14, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 205

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.3-36	3.3-36
3.3-65	3.3-65
3.3-68	3.3-68
3.5-6	3.5-6
3.5-10	3.5-10
3.6-2	3.6-2
3.6-40	3.6-40
3.8-18	3.8-18

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 205 TO

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated January 30, 2004 (ADAMS accession number ML040340740), as supplemented by letter dated June 17, 2004 (ADAMS accession number ML041740343), Nebraska Public Power District (the licensee), requested changes to the Technical Specifications (TSs) for Cooper Nuclear Station (CNS). The proposed changes would provide a one-time extension of various TS Surveillance Requirements (SR) listed below. The licensee is currently operating in Cycle 22. The next refueling outage (RFO) was originally scheduled in Fall 2004. However, to fully utilize the core the licensee plans to extend this current operating cycle to January 15, 2005. The extension will cause a limited number of SRs to expire beyond the 25 percent permitted by SR 3.0.2.

The proposed temporary note states:

> -------------------------------------TEMPORARY NOTE---The next required performance of this SR [ ] may be delayed until the current cycle refueling outage, but no later than February 2, 2005. This temporary note expires upon startup from that refueling outage.

A temporary note is being proposed for the following SRs:

1. SR 3.3.5.1.5, Logic System Functional Test, for Core Spray (CS) system initiation on low Reactor Vessel Water Level and high drywell pressure.

2. SR 3.3.8.1.2, Channel Calibration, for the following Loss of Power Instrumentation Functions:

    4.16 kV Emergency Bus Undervoltage (Loss of Voltage)
    4.16 kV Emergency Bus Normal Supply Undervoltage (Loss of Voltage)
    4.16 kV Emergency Bus Essential Station Service Transformer (ESST) Supply Undervoltage (Loss of Voltage)
    4.16 kV Emergency Bus Undervoltage (Degraded Voltage)
    4.16 kV Emergency Bus ESST Supply Undervoltage (Degraded Voltage);

3. SR 3.3.8.1.3, Logic System Functional Test, for the following Loss of Power Instrumentation Functions:

   4.16 kV Emergency Bus Undervoltage (Loss of Voltage)
   4.16 kV Emergency Bus Normal Supply Undervoltage (Loss of Voltage)
   4.16 kV Emergency Bus ESST Supply Undervoltage (Loss of Voltage)
   4.16 kVEmergency Bus Undervoltage (Degraded Voltage)
   4.16 kV Emergency Bus ESST Supply Undervoltage (Degraded Voltage)

4. SR 3.3.8.2.1, Channel Calibration for overvoltage, undervoltage, and underfrequency on the Reactor Protection System (RPS) Electric Power Monitoring System.

5. SR 3.3.8.2.2, System Functional Test on the RPS Electric Power Monitoring System.

6. SR 3.5.1.9, Verify each Emergency Core Cooling System (ECCS) injection/spray subsystem actuates on an actual or simulated automatic initiation signal. (Applicable Modes: 1, 2, and 3)

7. SR 3.5.2.5, Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal. (Applicable Modes: 4 and 5)

8. SR 3.6.1.1.2, Verify drywell to suppression chamber bypass leakage is equivalent to a hole less than 1.0 inch in diameter.

9. SR 3.6.4.3.2, Perform required Standby Gas Treatment (SGT) filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

   The VFTP, discussed in TS Section 5.5.7, consists of the following five parts:

   1. 5.5.7a. This part tests the high efficiency particulate air (HEPA) filter efficiency.
   2. 5.5.7b. This part tests charcoal adsorber in-place bypass and penetration.
   3. 5.5.7c. This part performs a laboratory test of charcoal samples.
   4. 5.5.7d. This part verifies that pressure drops across the filters is less than the maximum allowable.
   5. 5.5.7e. This part verifies that the SGT electric heaters dissipate the required amount of power.

10. SR 3.6.4.3.4, Verify the correct position of the SGT units cross tie damper, and that each SGT room air supply check valve and SGT dilution air shutoff valve can be opened.

11. SR 3.8.4.7, Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to a battery service test.

2.0 REGULATORY EVALUATION

The NRC staff finds that the licensee in Section 6.0 of its submittal identified the applicable regulatory requirements and guidance. The regulatory requirements and guidance for which the NRC staff based its acceptance are listed below.

Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that an applicant for a license authorizing operation of a production or utilization facility include proposed TS in its license application. The regulation at 10 CFR 50.36(c)(1) addresses safety limits, limiting safety system settings, and limiting control settings. The regulation at 10 CFR 50.36(c)(2) identifies limiting conditions for operation (LCOs) as a requirement for TSs. It defines an LCO as the lowest functional capability or performance level of equipment required for safe operation of the facility. The regulation at 10 CFR 50.36(c)(2)(ii) identifies four criteria for which an LCO is required if any one of the criteria is met. The regulation at 10 CFR 50.36(c)(3) identifies the SRs as one of the required parts of TS, and defines SRs as requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

The regulation at 10 CFR 50.46(a)(1)(i) requires that each boiling or pressurized light-water nuclear power reactor be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accident (LOCA) conforms to the criteria of 10 CFR 50.46(b). The five acceptance criteria for ECCS performance in nuclear power reactors are specified in 10 CFR 50.46(b).

Pursuant to 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” preventive maintenance activities must not reduce the overall availability of the systems, structures and components.

The regulation at 10 CFR 100.11 specifies limits (as reference values) for offsite doses following a postulated fission product release. These limits are a whole body dose of 25 rem and a total radiation dose to the thyroid from iodine exposure of 300 rem.

The CNS licensing basis requires conformance to Proposed General Design Criteria (GDC) 10, as reflected in Appendix F in the Updated Safety Analysis Report (USAR). Proposed GDC 10 states: “Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features [ESFs] as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.”

The regulation at 10 CFR Part 50, Appendix A, GDC 17, “Electric power systems,” requires, in part, that nuclear power plants have onsite and offsite electric power systems to perform the functioning of structures, systems, and components that are important to safety. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

According to 10 CFR Part 50, Appendix A, GDC 18, “Inspection and testing of electric power systems,” electric power systems that are important to safety must be designed to permit appropriate periodic inspections and testing.

Pursuant to 10 CFR Part 50, Appendix A, GDC 19, “Control room,” a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

TS Task Force (TSTF) Traveler TSTF-360, Revision 1, “DC Electrical Rewrite,” which was approved by the NRC staff on December 18, 2000, for incorporation into standard TS, provides guidance for the rewrite of current TS for Class 1E DC power supply systems.

3.0 TECHNICAL EVALUATION

The CS system is part of the CNS ECCS. The safety design bases of the ECCS includes providing adequate cooling of the reactor core under abnormal and accident conditions (i.e., LOCA) and removing heat to preserve the integrity of the fuel cladding.

The CS system is a low pressure system (i.e., designed to provide cooling to the core when reactor pressure is low) composed of two independent subsystems. Each subsystem consists of a motor driven-pump, a spray sparger above the core, and piping and valves to transfer water from the suppression pool to the sparger.

The CS system may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low (Level 1) or Drywell Pressure - High. Each of these diverse variables is monitored by four redundant switches, which are connected to relays. The relays send signals to two logic systems, with each system arranged in a one-out-of-two-taken-twice logic. Each logic system initiates one of the two CS pumps.

Upon receipt of an initiation signal, if normal AC power is available, both CS pumps start after an approximate 10-second time delay. If a core spray initiation signal is received when normal AC power is not available, the CS pumps start approximately 10 seconds after the bus is energized by the diesel generators (DGs).

The CNS critical loads are normally supplied from the main generator through a normal station service transformer. The offsite power source is supplied through a 161- 4.16/4.16 kV startup station service transformer from a 161 kV switchyard and a 345/161 kV, auto-transformer connected to a 345 kV switchyard, and a separate emergency bus ESST (emergency station service transformer) energized by a 69 kV line. The 161 kV switchyard has one incoming line and the 345 kV switchyard has five incoming lines. In addition, there are two onsite DGs to provide standby AC power.

If the normal station service transformer powered by the main generator is lost, the startup station service transformer, which is normally energized, will automatically energize the two 4160 Volt critical buses. If the startup station service transformer fails to provide power to the critical buses, the emergency station service transformer, which is normally energized, will automatically energize both the critical busses. If the emergency station service transformer also fails, then the DGs would automatically energize their respective buses. CNS also has two 125 Volt and two 250 Volt DC battery systems.

3.1 SR 3.3.5.1.5, Logic System Functional Test, for CS system initiation on low Reactor Vessel Water Level and high drywell pressure. Extension requested for 4 days for Subsystem A and 18 days for Subsystem B.

The Logic System Functional Test demonstrates the operability of the required initiation logic and simulates automatic actuation for a specific channel. SR 3.3.5.1.5 applies to more than one system or subsystem. The licensee's requested deferral applies only to performance of that part of the Logic System Functional Test for initiation of CS system (both subsystems A and B) on low reactor vessel water level and high drywell pressure to demonstrate that the injection valves open, the test return valve closes, and the pump breaker closes.

The last performance of this SR on CS subsystem A was on March 18, 2003, during the last refueling outage. With the 25 percent extension of the specified frequency allowed by SR 3.0.2, the next required performance of this SR is due no later than January 29, 2005. CNS plans to shutdown on January 15, 2005, to begin the RFO22. Division 1 will be the protected division during approximately the first half of the refueling outage. CS subsystem A is part of division 1. No maintenance or surveillance can be performed on a protected division. The requested one-time deferral to no later than February 2, 2005, is needed to allow time to complete maintenance or surveillance on Division 2 and make the changeover to Division 2 as the protected division, thereby allowing performance of that portion of this SR which demonstrates that the injection valves open, the test return valve closes, and the pump breaker closes on CS subsystem A.

The last performance of this SR on CS subsystem B was on March 4, 2003, during the last refueling outage. With the 25 percent extension of the specified frequency allowed by SR 3.0.2, the next required performance of this SR is due no later than January 15, 2005. The licensee plans to shutdown on January 15, 2005, to begin the RFO22. Division 1 will be the protected division during approximately the first half of RFO22. This SR on CS subsystem B will be performed during the first half of the outage. The licensee's requested one-time deferral to no later than February 2, 2005, is needed to allow time to perform that portion of the logic of this SR which demonstrates that the injection valves open, the test return valve closes, and the pump breaker closes.

The 18-month frequency is based on the need to perform the surveillance under the conditions that apply during a plant outage and to avoid the potential for an unplanned transient if the surveillance is performed with the reactor at-power.

The licensee's requested delay period for the performance of this surveillance is 4 days for CS Subsystem A and 18 days for CS Subsystem B. The portion of this SR to which this request applies uses an ECCS test switch to simulate a high drywell pressure/low reactor water level signal to the CS initiation logics. Each logic is tested to demonstrate the logic will cause the injection valves to open, the test return valve to close, and the pump breaker to close.

The delay in the surveillance interval of 4 days for CS Subsystem A, functions 1a and 1b, and 18 days for CS Subsystem B, functions 1a and 1b, will not impact the performance of the CS initiation logic. Based on the licensee's operating history of this portion of the logic, a 4-day and an 18-day extension of the CS initiation logic functional test are justified and will not impact operability of the system. The licensee has reviewed the three most recent surveillances based on the associated procedure and found no failures associated with this portion of the logic functional test. Therefore, based on the past performance, there is reasonable assurance that the proposed extension of 14 days over 22.5 months (18 months plus 25 percent) will have no impact on the ability to meet the license requirements.

3.1.1 Impact on Defense-in-Depth

The proposed change is needed to meet the following elements of the defense-in-depth principle:

3.1.1.1 A reasonable balance among preventing core damage, preventing containment failure, and consequence mitigation is preserved.

The proposed extension has only a small calculated impact on core damage frequency and large early release frequency. The change does not affect containment integrity. The change neither degrades core damage prevention at the expense of containment integrity, nor degrades containment integrity at the expense of core damage prevention. The balance between preventing core damage and preventing containment failure is the same. Consequence mitigation remains unaffected by the proposed changes. Furthermore, no new accident or transient is introduced with the requested change, and the likelihood of an accident or transient is not impacted. No new activities on the required initiation logic system will be performed at-power that could lead to a new transient event. Conversely, the increased surveillance test interval may reduce the likelihood of a test-induced transient or accident.

3.1.1.2 Over reliance on programmatic activities to compensate for weakness in plant design is avoided.

The plant design will not be changed to accommodate this SR interval extension. All safety systems will still function in the same manner with the same signals available to trip the reactor and initiate ESF functions, and there will be no additional reliance on additional systems, procedures, or operator actions. The calculated risk increase for the change is very small, and additional control processes are not required to compensate for any risk increase.

3.1.1.3 System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.

There is no impact on the redundancy, independence, or diversity of the required initiation logic system or of the ability of the plant to respond to events with diverse systems. This system is diverse and redundant and will remain so. There will be no change to the signals available to trip the reactor or initiate the ESF actuation system.

3.1.1.4 Defenses against potential common-cause failures are maintained and the potential for the introduction of new common-cause failure mechanisms is assessed.

Defenses against common-cause failures are maintained. The test interval extension is not sufficiently long to expect a new common-cause mechanism to occur. The operating environment for these components remains the same; therefore no new common-cause failure modes are expected. In addition, backup systems and operator actions are not impacted by the change; and there are no common-cause links between the required logic initiation system and these backup options.

3.1.1.5 Independence of barriers is maintained.

The barriers protecting the public and the independence of these barriers are maintained. It is not expected that multiple systems will be out of service simultaneously, which could lead to degradation of these barriers and an increase in risk to the public with implementation of the extended slave relay test interval.

3.1.1.6 Defenses against human errors maintained.

No new operator actions related to the slave relay test interval extension are required. No additional operation or maintenance procedures have been introduced nor are existing procedures required to be revised due to the change. No new at-power test or maintenance activities are expected to occur as a result of the change. Surveillance tests will not be performed at-power, which will reduce the potential for test-induced reactor trips and safety system actuations. This represents a risk benefit, i.e., a reduction in risk.

3.1.2 Impact on Safety Margins

The safety analysis acceptance criteria as stated in the USAR are not impacted by the proposed change. Diversity with regard to the signals, which provide a reactor trip and actuation of ESFs, will also be maintained. The proposed change will not result in operation of the unit in a configuration outside of the design basis. All signals credited as primary or secondary and all operator actions credited in the accident analysis will remain the same. Therefore, there is no impact on the safety margins.

Based on the above evaluation, the NRC staff concludes that this requested one-time deferral of the SRs does not adversely impact the ability of the safety system. The licensee continues to meet the requirements of 10 CFR 50.36, to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

3.2 SR 3.3.8.1.2, Channel Calibration, for the following Division 1 Loss of Power Instrumentation Functions. Extensions requested for 1 day for Functions 1a, 1b, 2a, 2b, 3a, and 3b and 3 days for Functions 4a, 4b, 4c, 5a, and 5b.

4.16 kV Emergency Bus Undervoltage (Loss of Voltage)
4.16 kV Emergency Bus Normal Supply Undervoltage (Loss of Voltage)
4.16 kV Emergency Bus ESST Supply Undervoltage (Loss of Voltage)
4.16 kV Emergency Bus Undervoltage (Degraded Voltage)
4.16 kV Emergency Bus ESST Supply Undervoltage (Degraded Voltage);

The licensee stated that the three most recent performances of this SR demonstrated that the as-found settings of the applicable relays were within the range specified in the procedures. Furthermore, the licensee provided setpoint calculation results for functions 4a, 4b, and 5a for a conservative 24-month calibration interval that indicated that the maximum expected instrument drift will be within the desired level of instrument Allowable Values. By letter dated June 17, 2004, in response to a staff request for additional information, the licensee confirmed that the evaluation of the 24-month Allowable Values were based on a linear extrapolation of the 18-month Allowable Value. The performance of channel functional test on these circuits to SR 3.3.8.1.1 every 31 days ensures operability of these channels. The channel functional test for functions 4c and 5b includes a calibration check of the instrument settings. This extension of calibration interval apply to Division 1 instrumentation only and the other Division will be unaffected by this change. The requested 1 or 3 days extension for this SR is significantly less than the SR calibration interval of 18 months (22.5 months given 25 percent margin). Based on the above considerations, the NRC staff concludes the proposed TS change is acceptable.

3.3 SR 3.3.8.1.3, Logic System Functional Test, for the following Division 1 Loss of Power Instrumentation Functions. Extension requested for 1 day for Functions 1a, 1b, 2a, 2b, 3a, and 3b and 3 days for Functions 4a, 4b, 4c, 5a, and 5b.

4.16 kV Emergency Bus Undervoltage (Loss of Voltage)
4.16 kV Emergency Bus Normal Supply Undervoltage (Loss of Voltage)
4.16 kV Emergency Bus ESST Supply Undervoltage (Loss of Voltage)
4.16 kV Emergency Bus Undervoltage (Degraded Voltage)
4.16 kV Emergency Bus ESST Supply Undervoltage (Degraded Voltage)

The three most recent performances of the channel calibration demonstrated that the as-found settings of these relays were within the TS limits. Performance of channel functional tests to SR 3.3.8.1.1 within every 31 days assures operability of these channels. This extension of calibration interval apply to Division 1 instrumentation only and the other Division will be unaffected by this change. Furthermore, the requested 1- or 3-day extensions for this SR are significantly less than the SR test interval of 18 months (22.5 months given 25 percent margin). Based on the above considerations, the NRC staff concludes the proposed TS change is acceptable.

3.4 SR 3.3.8.2.1, Channel Calibration for overvoltage, undervoltage, and underfrequency on the RPS Electric Power Monitoring System, Division 1. Extension requested for 1 day.

This TS change for extension of overvoltage, undervoltage, and underfrequency SRs on the RPS Electric Power Monitoring system by one day is only for Division 1 instrumentation and the other Division will be unaffected by this change. The licensee stated that the most three recent performances of this surveillance have shown that the as-found values for the channels are within the range specified in the procedures except for two channels which were found out of tolerance and were adjusted within tolerance during RFO19 and another channel was adjusted during RFO20. Since then, the as-found values of the channels had minimum variance from the as-left values from the previous performance. The licensee also stated that calculation shows an estimated monthly drift of only 0.017 Hertz (Hz) which is comparable to their data rounding value of 0.02 Hz. Furthermore, the requested 1-day extension for this SR is significantly less than the SR calibration interval of 18 months (22.5 months given 25 percent margin). Based on the above considerations, the NRC staff concludes the proposed TS change is acceptable.

3.5 SR 3.3.8.2.2, System Functional Test on the RPS Electric Power Monitoring System, Division 1. Extension requested for 1 day.

The RPS electric power monitoring system has two assemblies in series located in a climate controlled area not subjected to the extremes in temperature or humidity. The three most recent performances of this SR on Division 1 demonstrated that the monitoring system functioned within TS limits. This extension of calibration interval is for Division 1 instrumentation only and the other Division will be unaffected by this change. Based on the above considerations, the NRC staff concludes the proposed TS change is acceptable.

3.6 SR 3.5.1.9, verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal. (Applicable Modes: 1, 2, and 3) and

SR 3.5.2.5, verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal (Applicable Modes: 4 and 5). Extension requested for 4 days for Subsystem A and 18 days for Subsystem B.

Both TSs SR 3.5.1.9 and SR 3.5.2.5 require verification that each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal every 18 months. However, SR 3.5.1.9 is applicable in Modes 1, 2, and 3 while SR 3.5.2.5 is applicable in Modes 4 and 5. The requested deferral applies to the portion of the surveillance on the CS system that demonstrates the logic system that causes the injection valves to open, the test return valves to close, and the pump breaker to close. It applies to both Subsystems A and B.

The last surveillance performance of SR 3.5.1.9 and SR 3.5.2.5 on CS Subsystem A was on March 18, 2003, during the last RFO (Cycle 21). SR 3.0.2 allows for a 25 percent extension of the interval specified in the frequency. With this extension, the SR 3.5.1.9 and SR 3.5.2.5 must be performed by January 29, 2005. The licensee plans to shutdown on January 15, 2005, in order to begin RFO22. During the first half of RFO22, maintenance and surveillances will be performed on Division 2 and Division 1 will be the protected division. CS Subsystem A is part of Division 1. No maintenance or surveillance can be performed on a protected division. When the maintenance and surveillances have been completed on Division 2, the maintenance and surveillances will be performed on Division 1 and the transfer will be made to Division 2 as the protected division. The requested deferral to February 2, 2005, will result in a delay period for these surveillances on CS Subsystem A of four days.

The last surveillance performance of SR 3.5.1.9 and SR 3.5.2.5 on CS Subsystem B was on March 4, 2003, during the last RFO (Cycle 21). With the 25 percent extension allowed by SR 3.0.2, the SR 3.5.1.9 and SR 3.5.2.5 must be performed on Division 2 by January 15, 2005. The maintenance and surveillances will be performed on Division 2 during the first half of RFO22. CS Subsystem B is part of Division 2. The requested deferral to February 2, 2005, will result in a delay period for these surveillances on CS Subsystem B of 18 days.

The delay periods of 4 days for CS Subsystem A and 18 days for CS Subsystem B apply to the parts of SR 3.5.1.9 and SR 3.5.2.5 that require verification for each ECCS injection/spray system to actuate on a simulated automatic initiation signal. An ECCS test switch is used to simulate a Level 1 reactor low water level and high drywell pressure signal to the CS initiation logic. Performance of the surveillances verifies the starting logic for the CS initiation logic. The logic is tested to verify that the logic will cause the injection valves to open, the test return valve to close, and the pump breaker to close. During the three most recent performances of these surveillances on CS Subsystem A and CS Subsystem B, the CS initiation logic functioned properly.

The due date for the performance of SR 3.5.1.9 for CS Subsystem A is January 29, 2005. However, SR 3.5.1.9 is only applicable in Modes 1, 2, and 3. The licensee plans to shutdown on

January 15, 2005. The shutdown will take the plant from Modes 1, 2, and 3 to Modes 4 and 5. The plant will be in Mode 4 or 5 on the due date for the performance of the surveillance. Therefore, the surveillance will not be applicable on this due date. The surveillance will be performed on CS Subsystem A during second half of the RFO. This performance will occur before the licensee startups and enters into Modes 1, 2, and 3.

The due date for the performance of SR 3.5.1.9 for CS Subsystem B is January 15, 2005. However, SR 3.5.1.9 is only applicable in Modes 1, 2, and 3. The shutdown of CNS on January 15, 2005, will take the plant from Modes 1, 2, and 3 to Modes 4 and 5. The plant will only be in Modes 1, 2, and 3 for a very short period during the process of shutting down before it enters Modes 4 and 5 on January 15, 2005. During this period, the low pressure coolant injection (LPCI) system of the Residual Heat Removal system is available. The LPCI system starts from the same signals (high drywell pressure or Level 1 reactor vessel water level) as the CS system to achieve the same objective of cooling the reactor core following a LOCA. The LPCI system acts independently through a different flowpath than the CS system to flood the reactor vessel. Since the period is very short and the LPCI system is available, the probability of core damage is low.

The performance of SR 3.5.2.5 is applicable in Modes 4 and 5. The plant will enter these modes when it shutdowns on January 15, 2005. The surveillance will be applicable during this shutdown. As mentioned above, there will be a delay in the performance of SR 3.5.2.5 on CS Subsystems A and B. During this delay, the LPCI system will be available to backup the CS system. Due to the availability of the LPCI system, the probability of core damage is low.

The licensee has requested a short, one-time extension of SR 3.5.1.9 and SR 3.5.2.5 that results in a delay in the surveillance interval of 4 days for CS Subsystem A and of 18 days for CS Subsystem B. The NRC staff reviewed the licensee’s request and noted that the CS initiation logic operated properly during the three most recent performances for the surveillances. The NRC staff concludes that the increased one-time extension is acceptable because it does not significantly increase the possibility for failure of the CS system.

Based on the above, the NRC staff concludes that the one-time extension of SR 3.5.1.9 and SR 3.5.2.5 of 4 days for CS Subsystem A and of 18 days for Subsystem B does not significantly increase the probability that a failure will occur in any of the related equipment covered by these TSs. The proposed one-time extensions of SR 3.5.1.9 and SR 3.5.2.5 do not adversely affect the ability of the CS system to continue to satisfy the criteria of 10 CFR 50.46 and 10 CFR 50.36(c)(3), therefore the NRC staff finds these delays acceptable.

3.7 SR 3.6.1.1.2, verify drywell to suppression chamber bypass leakage is equivalent to a hole less than 1.0 inch in diameter. Extension requested for 26 days.

The licensee stated and the NRC staff agrees that the function of the primary containment is to isolate and contain fission products released from the Reactor Primary system following a design basis LOCA and to confine the postulated release of radioactive material. The primary containment consists of a steel pressure vessel in the shape of an inverted light bulb with a torus-shape suppression chamber located below and encircling the drywell. The drywell surrounds the Reactor Primary system and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment. Maintaining the pressure suppression function of the primary containment requires limiting the leakage from the drywell to the suppression chamber. If an event were to occur that pressurize the drywell, the steam would be directed through the downcomers into the suppression pool.

SR 3.6.1.1.2 is a leak test that confirms that the bypass area between the drywell and the suppression chamber is less than a one-inch diameter hole. This ensures that the leakage paths that would bypass the suppression pool are within allowable limits.

The licensee stated that SR 3.6.1.1.2 was last performed on February 25, 2003, during the RFO21. With the 25 percent extension allowed by SR 3.0.2 the next required performance is due January 7, 2005. The licensee is requesting a deferral to February 2, 2005. The deferral is based on the need to perform this SR during RFO22.

In order to determine if the requested extension could be granted and reasonable assurance of safety maintain, the NRC staff examined the surveillance and maintenance history for this item that was provided to the NRC staff via the licensee's submittal. The licensee stated that an excess leak rate from the torus to drywell vacuum breakers was experienced during a drywell to torus leak test on May 17, 1997. Corrective action was taken at the time and leak testing was successfully completed on May 18, 1997, at the end of RFO17. Leak testing was again completed satisfactory on September 28, 1998, and December 14, 1998, during Cycle RFO18. In March 2000 another failure was experienced during the leak test conducted during RFO19. In that outage more extensive corrective actions were taken to improve the overall vacuum breaker performance including the leak tightness of the vacuum breakers. Improved maintenance training and new procedural guidance for testing and maintenance have resulted in significantly improved leak test performance. Since implementation of the corrective actions in RFO19, successful leak testing has been performed on April 16, 2000, November 4 and December 27, 2001, and February 25, 2003, with no failures experienced. The total leak rate has consistently been a small fraction of allowable limits.

While examining this information, the NRC staff specifically focused on the low leakage rates observed since RFO19, the relative short period of 26 days for the SR extension, and the low probability of an accident within those 26 days. As a result of the NRC staff's examination and qualitative analysis, it is the NRC staff's judgement that the requested extension should be granted, and that reasonable assurance of safety will be maintained. Therefore, the NRC staff finds it acceptable to grant the requested extension.

3.8 SR 3.6.4.3.2, perform required SGT filter testing in accordance with the VFTP. The VFTP, discussed in TS Section 5.5.7, consists of the following five parts, with extensions requested for 14 to 16 days:

1. 5.5.7a. This part tests the HEPA filter efficiency.
2. 5.5.7b. This part tests charcoal adsorber in-place bypass and penetration.
3. 5.5.7c. This part performs a laboratory test of charcoal samples.
4. 5.5.7d. This part verifies that pressure drops across the filters is less than the maximum allowable.
5. 5.5.7e. This part verifies that the SGT electric heaters dissipate the required amount of power.

SR 3.6.4.3.2 verifies that the required SGT filter testing is performed in accordance with the VFTP. The VFTP includes testing the HEPA filter's performance, charcoal adsorber efficiency, minium system flow rate, and some physical properties of the activated charcoal. The applicable section of the VFTP is TS Section 5.5.7.

In its submittal, the licensee stated that portions of SR 3.6.4.3.2 on SGT Train A were last performed on March 7 and 8, 2003. With the 25 percent extension allowed by SR 3.0.2 the next required performance is due on January 18 and 19, 2005. Portions of this SR on SGT Train B were last performed on March 6 and 7, 2003. With the 25 percent extension allowed by SR 3.0.2 the next required performance is due on January 17 and 18, 2005. The licensee is requesting a deferral to February 2, 2005, based on the need to schedule this SR during RFO22.

In it's review and analysis, the NRC staff considered the following information provided by the licensee.

The licensee stated that the requested delay period for any part of the VFTP for either Train A or Train B is a maximum of 16 days. And, that the following are the specific parts of the VFTP and the basis for the requested extension for each.

1. TS 5.5.7a. This part tests the HEPA filter efficiency. SGT HEPA filter efficiency is trending well and has ample margin to the minimum. Filter efficiency decreases very slowly over time.

2. TS 5.5.7b. This part tests charcoal adsorber in-place bypass and penetration. This is trending well, with ample margin, and is not subject to abrupt decreases in efficiency.

3. TS 5.5.7c. This part performs a laboratory test of charcoal samples. SGT charcoal efficiency has ample margin to the required minimum and decreases very slowly over time.

4. TS 5.5.7d. This part verifies that pressure drops across the filter are less than the maximum allowable. SGT filter pressure drops are well below the allowable maximum. Pressure drop increases very slowly as dust loading increases.

5. TS 5.5.7e. This part verifies that the SGT electric heaters dissipate the required amount of power. The SGT heater power has ample margin to the required minimum and show a stable trend.

The licensee also stated that the three most recent performances of the surveillances for SGT filter testing in accordance with the VFTP were reviewed. This review determined that no failures had been experienced. The NRC staff's review considered the information outlined above, however, the NRC staff's conclusions are based on two considerations only. These two considerations are the number of days involved in the requested extension and the record of no failures over the three most recent performances of the SRs. It is the NRC staff's judgement that the requested 16 days extension is not unreasonable when considered concurrent with no failures experienced over the three most recent performances of the SR. Therefore the NRC staff finds it acceptable to grant the requested extension.

3.9 SR 3.6.4.3.4, verify the correct position of the SGT units cross tie damper, and that each SGT room air supply check valve and SGT dilution air shutoff valve can be opened. Extension requested for 14 days for Train A and 15 days for Train B.

SR 3.6.4.3.4 verifies that the SGT units cross tie damper is in the correct position, and that each SGT room air supply check valve and each air operated SGT dilution air shutoff valve open when required. This ensures that the decay heat removal function of SGT system operation is available. The licensee stated: "[o]perating experience has shown that these components will pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was found to be acceptable from a reliability standpoint."

The licensee also stated that SR 3.6.4.3.4 was last performed on SGT Train A on March 8, 2003. With the 25 percent extension allowed by SR 3.0.2 the next required performance is due on January 19, 2005. This SR was last performed on SGT Train B on March 7, 2003. With the 25 percent extension allowed by SR 3.0.2 the next required performance is due on January 18, 2005. Again, the deferral to February 2, 2005, is based on the need to schedule this SR during RFO22.

In support of the request for a SR extension the licensee stated the following:

> The requested delay period for the performance of SR 3.6.4.3.4 is 14 days for SGT Train A and 15 days for Train B.
>
> The three most recent performances of this surveillance on the SGT System were reviewed. CNS has not experienced failures of the room air dilution valves or the check valves to open when required. A surveillance is performed every 18 months that measures cooling air flow through the cross tie damper to ensure sufficient cooling flow for the decay heat removal function. Cross tie flow has historically varied only slightly, with ample margin to the minimum and maximum required flow. However, the review of the surveillances identified that one failure of the cooling flow being below the minimum acceptance criteria had occurred. The crosstie damper was adjusted and the surveillance was successful on the repeat performance. The crosstie damper is a manual valve locked in the throttled position, and would not be expected to cause abrupt, significant changes in the cross tie flow.

The NRC staff reviewed this information with particular attention on the historical performance and the length of the requested extension. The performance of the components in question over the most recent surveillance resulted in no failures. The historical performance (reliability) indicates that failure is not likely in the requested extension period of 15 days. In addition the NRC staff also considered the required function of the components in question and concluded that it is reasonable to expect that these components will perform as required in the unlikely event of a design basis event occurring during the 15 days extension period. Therefore the NRC staff finds the extension request acceptable.

3.10 SR 3.8.4.7, verify battery capacity is adequate to supply and maintain in operable status, the required emergency loads for the design duty cycle when subjected to a battery service test. Extensions requested for 1 day for Division 1 125-Volt Battery and 5 days for Division 1 250-Volt Battery.

125 Volt Battery

The licensee stated that the latest load profile calculation based on the Institute of Electrical and Electronics Engineers' (IEEE) 485 battery sizing technique, shows that this battery meets the design requirement by 127 percent including a 15 percent design margin, a 90 percent aging factor, and a temperature correction factor to the minimum allowable of 70 degrees Fahrenheit (°F). Further, this battery was replaced in November of 2001 and was subjected to a modified performance test which showed a capacity of approximately 97 percent of factory rating. Since then, no new load has been added to the battery system. Furthermore, the requested 1-day extension for this SR is significantly less than the surveillance interval of 18 months (22.5 months given 25 percent margin). In addition, this extension of calibration interval is for Division 1 instrumentation only and the other Division will be unaffected by this change. Based on the above considerations, the NRC staff concludes that the proposed TS change is acceptable.

250 Volt Battery

The licensee stated that the latest load profile calculation based on IEEE 485 battery sizing technique, shows this battery meets the design requirement by approximately 192 percent including a 15 percent design margin, a 90 percent aging factor, and a temperature correction factor to the minimum allowable of 70 °F. Further, this battery with expected life of 20 years was replaced in 1995 and was subjected to a modified performance test in December 2001, which showed a capacity of approximately 92.5 percent of factory rating. No new load has been added to the battery system.

On August 25, 1999, C&D Technologies, the manufacturer of the 125 Volt and 250 Volt batteries, informed the licensee by a 10 CFR Part 21 Notice about the possibility of positive plate growth from high levels of calcium within the positive plate grids of the batteries. The 125 Volt batteries have been replaced as the resolution of this Part 21 Notice. For the 250 Volt batteries, the licensee has initiated a semi-annual visual inspection procedure under their Corrective Action Program until all battery cells subject to the Part 21 Notice are removed from the plant. The requested 5-day extension for this surveillance interval is significantly less than the SR interval of 18 months (22.5 months given 25 percent margin). In addition, this extension of calibration interval is for Division 1 instrumentation only and the other Division will be unaffected by this change. Based on the above considerations, the NRC staff concludes that the proposed TS change is acceptable.

3.11 Technical Conclusions

As indicated above, the NRC staff reviewed the licensee's request to add a temporary note to certain TS SRs. The temporary note will state that the next required performance of the applicable SR may be delayed until the current cycle RFO, but no later than February 2, 2005. The note will also state that it expires upon startup from the RFO to emphasize its temporary nature. The NRC staff concluded that with the one-time delay in the performance of the SRs for short periods as outlined above, provides reasonable assurance that safety is maintained, and that the requested temporary change to the TSs is acceptable.

The licensee evaluated the risk impact of extension of the SR for the automatic initiation of the CS subsystem by increasing the failure-to-start probabilities of the CS A and B pumps and calculated that the incremental changes to the core damage probability and large early release probability were not risk significant. Additionally, the licensee provided a cumulative increase of incremental core damage probability due to all SR extensions, which is beyond the scope of work of this evaluation. This information was not considered essential to the NRC staff's review and was not reviewed by the NRC staff.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published February 12, 2004 (69 FR 7023). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: B. Grimmel
H. Walker
S. Mazumdar
S. Rhow

Date: July 14, 2004

cc:

Mr. William J. Fehrman
President and Chief Executive Officer
Nebraska Public Power District
1414 15th Street
Columbus, NE 68601

Mr. Clay C. Warren
Vice President of Strategic Programs
Nebraska Public Power District
1414 15th Street
Columbus, NE 68601

Mr. John R. McPhail, General Counsel
Nebraska Public Power District
P. O. Box 499
Columbus, NE 68602-0499

Mr. Paul V. Fleming
Licensing Manager
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

Mr. Michael J. Linder, Director
Nebraska Department of Environmental Quality
P. O. Box 98922
Lincoln, NE 68509-8922

Chairman
Nemaha County Board of Commissioners
Nemaha County Courthouse
1824 N Street
Auburn, NE 68305

Ms. Cheryl K. Rogers, Program Manager
Nebraska Health & Human Services System
Division of Public Health Assurance
Consumer Services Section
301 Centennial Mall, South
P. O. Box 95007
Lincoln, NE 68509-5007

Mr. Ronald A. Kucera, Director of Intergovernmental Cooperation
Department of Natural Resources
P.O. Box 176
Jefferson City, MO 65102

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 218
Brownville, NE 68321

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Jerry Uhlmann, Director
State Emergency Management Agency
P. O. Box 116
Jefferson City, MO 65101

Chief, Radiation and Asbestos Control Section
Kansas Department of Health and Environment
Bureau of Air and Radiation
1000 SW Jackson
Suite 310
Topeka, KS 66612-1366

Mr. Daniel K. McGhee
Bureau of Radiological Health
Iowa Department of Public Health
401 SW 7th Street
Suite D
Des Moines, IA 50309

Mr. Scott Clardy, Director
Section for Environmental Public Health
P.O. Box 570
Jefferson City, MO 65102-0570

Jerry C. Roberts, Director of Nuclear
Safety Assurance
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321