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NLS2008019
March 6, 2008

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

Subject: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications – Appendix K Measurement Uncertainty Recapture Power Uprate
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:
1. Letter from Carl F. Lyon, U.S. Nuclear Regulatory Commission, to Stewart B. Minahan, Nebraska Public Power District, dated January 23, 2008, "Cooper Nuclear Station – Request for Additional Information RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MD7385)"
2. Letter from Carl F. Lyon, U.S. Nuclear Regulatory Commission, to Stewart B. Minahan, Nebraska Public Power District, dated February 4, 2008, "Cooper Nuclear Station – Request for Additional Information RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MD7385)"
3. Letter from Stewart B. Minahan, Nebraska Public Power District, to the U.S. Nuclear Regulatory Commission, dated November 19, 2007, "License Amendment Request to Revise Technical Specifications - Appendix K Measurement Uncertainty Recapture Power Uprate"

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District (NPPD) to submit a response to the Nuclear Regulatory Commission (NRC) Request for Additional Information sent on January 23, 2008 (Reference 1) and February 4, 2008 (Reference 2). The additional information requested is to support NRC review of the license amendment request (LAR) to revise the Cooper Nuclear Station (CNS) Technical Specifications for Measurement Uncertainty Recapture power uprate. This LAR was submitted by NPPD letter dated November 19, 2007 (Reference 3).

Attachment 1 contains a response to Reference 1. However, the response to NRC Question I.2 is not available at this time. The response to this question will be provided in a separate correspondence no later than March 13, 2008. Attachment 2 contains a response to Reference 2.

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Attachment 3 contains a final, for-information, copy of Technical Requirements Manual Page 3.3-22. None of the attachments contain information considered proprietary as defined by 10 CFR 2.390.

The information submitted by this letter (including attachments) does not change the conclusion of the No Significant Hazards Consideration evaluation submitted by the Reference 3 letter. Should you have any questions regarding this submittal, please contact David Van Der Kamp, Licensing Manager, at (402) 825-2904.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on MAR 6 2008

Sincerely,

Stewart B. Minahan

Stewart B. Minahan
Vice President - Nuclear and
Chief Nuclear Officer

/dm

Attachments

cc: Regional Administrator w/ attachments
USNRC - Region IV

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

Senior Resident Inspector w/ attachments
USNRC - CNS

Nebraska Health and Human Services w/ attachments
Department of Regulation and Licensure

NPG Distribution w/o attachments

CNS Records w/ attachments

Attachment 1
Response to Request for Additional Information (RAI),
Dated January 23, 2008,
Regarding License Amendment Request (LAR) to Revise Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate
Cooper Nuclear Station (CNS), Docket No. 50-298, DPR-46

The Nuclear Regulatory Commission (NRC) RAIs are shown in italics and Nebraska Public Power District's (NPPD) response shown in block font.

NRC Request

I. The following questions are provided from the Steam Generator and Chemical Engineering Branch (CSGB):

1. The flow accelerated corrosion (FAC) monitoring program includes the use of a predictive method to calculate the wall thinning of components susceptible to FAC. In order for the U.S. Nuclear Regulatory Commission (NRC) staff to evaluate the accuracy of these predictions, the staff requests a sample list of components for which wall thinning is predicted and measured by ultrasonic testing or other methods. Include the initial wall thickness (nominal), current (measured) wall thickness, and a comparison of the measured wall thickness to the thickness predicted by the model.

NPPD Response

A sample list of components that were inspected during the most recently completed refueling outage is provided in Table 1. The list includes components from four different systems (Extraction Steam, Condensate, Condensate Drain, and Feedwater). Components are included from the Extraction Steam piping to the third Feedwater (FW) Heater, which is predicted to have the greatest increase in wear as a result of the power uprate. The list includes the nominal, predicted, and actual thickness as well as the difference between the actual and predicted thicknesses (all dimensions are in inches). As can be seen in the information provided, the model has yielded results that show the actual measured wall thicknesses were greater than those predicted by the model.

Table 1 – Comparison of Predicted versus Actual Wall Thickness*

Component ID	System	Size	Nominal Thickness	Predicted Thickness	Measured Thickness (RE23)	Actual - Predicted
BS-E-15-2812-2	Ex. Steam to FWH#2	24	0.500	0.365	0.428	0.063
BS-E-17-2812-2	Ex. Steam to FWH#2	24	0.500	0.438	0.451	0.013
BS-E-19-2812-2	Ex. Steam to FWH#2	24	0.500	0.365	0.456	0.091
BS-E-3-EC93877SP-1A	Ex. Steam to FWH#2	24	0.375	0.269	0.367	0.098
BS-E-10-EC93877SP-1A	Ex. Steam to FWH#3	20	0.375	0.224	0.312	0.088
BS-E-12-EC93877SP-1B	Ex. Steam to FWH#3	20	0.375	0.251	0.287	0.036
BS-E-14-EC93877SP-1A	Ex. Steam to FWH#3	20	0.375	0.247	0.289	0.042
BS-E-2-2812-1	Ex. Steam to FWH#3	20	0.375	0.258	0.294	0.036
BS-E-3-2812-2	Ex. Steam to FWH#3	20	0.375	0.238	0.312	0.074
BS-E-4-2812-1	Ex. Steam to FWH#3	20	0.375	0.238	0.281	0.043
BS-E-5-2812-1	Ex. Steam to FWH#3	20	0.375	0.186	0.311	0.125
BS-E-6-2812-2	Ex. Steam to FWH#3	20	0.375	0.288	0.304	0.016
BS-E-7-2812-1	Ex. Steam to FWH#3	20	0.375	0.251	0.252	0.001
BS-N-2-2812-2	Ex. Steam to FWH#3	20	0.375	0.166	0.312	0.146
BS-P-8-EC93877SP-1B	Ex. Steam to FWH#3	20	0.375	0.232	0.287	0.055
CH-E-18-2819-3	Cond. FWH#3 to FWH#4	16	0.500	0.236	0.445	0.209
CH-R-5-2819-3	Cond. FWH#3 to FWH#4	18 X 16	0.562	0.403	0.539	0.136
CH-E-10-2819-6	Cond. FWH#4 to FWH#5	18	0.562	0.51	0.601	0.091
CH-E-4-2819-4	Cond. FWH#4 to FWH#5	16	0.500	0.355	0.434	0.079
CH-E-7-2819-4	Cond. FWH#4 to FWH#5	16	0.500	0.295	0.478	0.183
CH-E-7-2819-6	Cond. FWH#4 to FWH#5	16	0.500	0.353	0.413	0.060
CH-E-8-2819-4	Cond. FWH#4 to FWH#5	16	0.500	0.535	0.652	0.117
CH-R-2-2819-6	Cond. FWH#4 to FWH#5	18 X 16	0.562	0.425	0.484	0.059
DR-T-5-2827-2	Moisture Separator Drain	12 X 8	0.375	0.367	0.456	0.089
DR-T-5-2827-4	Moisture Separator Drain	12 X 8	0.375	0.301	0.397	0.096
RF-E-12-2849-4	Reactor Feedwater	18	1.375	1.344	1.381	0.037
RF-E-17-2849-4	Reactor Feedwater	18	1.375	1.429	1.432	0.003
RF-E-19-2849-4	Reactor Feedwater	24	1.812	1.773	1.783	0.010
RF-E-21-2849-4	Reactor Feedwater	18	1.375	1.336	1.376	0.040
RF-E-4-2509-2	Reactor Feedwater	12	1.125	0.992	1.095	0.103
RF-E-5-2509-2	Reactor Feedwater	12	1.125	0.867	0.891	0.024
RF-E-6-2849-4	Reactor Feedwater	18	1.375	1.137	1.361	0.224
RF-E-7-2849-4	Reactor Feedwater	18	1.375	1.348	1.394	0.046
RF-E-8-2509-1	Reactor Feedwater	12	1.125	0.867	1.109	0.242
RF-E-8-2509-2	Reactor Feedwater	12	1.125	0.996	1.047	0.051
RF-N-1-2849-4	Reactor Feedwater	18 X 20	1.330	1.115	1.167	0.052
RF-N-2-2849-4	Reactor Feedwater	18 X 20	1.330	1.117	1.134	0.017
RF-O-1-2849-4	Reactor Feedwater	18	1.375	1.128	1.239	0.111

Component ID	System	Size	Nominal Thickness	Predicted Thickness	Measured Thickness (RE23)	Actual - Predicted
RF-P-14-2849-4	Reactor Feedwater	18	1.375	1.263	1.295	0.032
RF-R-1-2509-2	Reactor Feedwater	18 X 12	1.562	1.59	1.651	0.061

*All values taken from CHECWORKS SFA predictive model for CNS.

NRC Request

2. *The power uprate will affect several process variables that influence FAC. Identify the systems that are expected to experience the greatest increase in wear as a result of the power uprate and discuss the effect of individual process variables (i.e., moisture content, temperature, oxygen, and flow velocity) on each system identified. For the most susceptible systems and components, what is the total predicted increase in wear rate due to FAC as a result of power uprate conditions?*

NPPD Response

NPPD has not yet completed the response to this question. The completed response to NRC Question I.2 will be provided in a separate correspondence to the NRC no later than March 13, 2008.

NRC Request

II. The following question is provided from the Vessels and Internals Integrity Branch (CVIB):

1. *Table 3-1 of Enclosure 1 to the submittal reported the peak end-of-license, i.e., 32 effective full power years (EFPY), reactor vessel (RV) inside diameter (ID) fluence, considering the measurement uncertainty recapture (MUR) power uprate, as 1.68×10^{18} n/cm^2 (E>1.0 MeV) for lower-intermediate shell plates and all welds. Based on this, the staff estimated that the ID fluence for 30 EFPY is 1.575×10^{18} n/cm^2 (E>1.0 MeV) [1.68x30/32]. The current Cooper Nuclear Station (CNS) technical specifications contain pressure-temperature (P-T) limit curves valid for 30 EFPY based on a projected peak RV ID fluence of 1.57×10^{18} n/cm^2 (E>1.0 MeV) for the limiting beltline material (the lower-intermediate longitudinal weld), as evaluated in the safety evaluation dated January 24, 2006. Please explain the need to revise the CNS P-T limit curves to 28 EFPY.*

NPPD Response

The fluence for Thermal Power Optimization (TPO) was calculated by scaling the Extended Power Uprate Effective Full Power Years (EFPY) (24) by 101.7%.

Peak Surface Fluence:
Current Licensed Thermal Power (CLTP) fluence (32 EFPY) = 1.67E18 n/cm^2 (TransWare calculation)
CLTP fluence (24 EFPY) = (1.67E18)/32 * 24 = 1.25E18 n/cm^2
TPO fluence (32 EFPY) = 1.017 * 1.67E18 n/cm^2 = 1.70E18 n/cm^2
TPO fluence (32-24 EFPY) = (1.70E18)/32 * (32-24) = 4.25E17 n/cm^2
Total CLTP + TPO fluence (32 EFPY) = 1.25E18 + 4.25E17 = 1.675E18 n/cm^2
Rounding up = 1.68E18 n/cm^2

Since the maximum beltline fluence calculated above is slightly higher than the fluence used in the generation of the current Pressure – Temperature (P-T) curves, the P-T curves should be limited to 28 EFPY: [(1.57E18 n/cm^2)/(1.68E18 n/cm^2)* 32 EFPY= 29.9 EFPY].

NRC Request

III. The following questions are provided from the Fire Protection Branch (AFPB):

1. *The staff notes that the General Electric Company (GE)-Hitachi Safety Analysis Report for CNS Thermal Power Optimization (TPO), NEDC-33385P, Revision 0, November 2007, Section 6.7, "Fire Protection," states that operation of the plant at the TPO level does not affect fire detection and suppression systems. Please address the impact of TPO uprate conditions on other fire protection program elements. At a minimum, include the following: (1) administrative controls, (2) fire barriers, (3) fire protection responsibilities of plant personnel, and (4) procedures and resources necessary for systems required to achieve and maintain safe-shutdown.*

NPPD Response

A review was conducted of the Fire Protection Program as related to administrative controls, fire barriers, fire protection responsibilities of plant personnel and resources necessary for systems required to achieve and maintain safe-shutdown. The review looked at the impact of TPO uprate and how it would impact these areas. The TPO uprate will have no impact on fire protection administrative controls, fire barriers, fire protection responsibilities of plant personnel, or resources necessary for systems required to achieve and maintain safe-shutdown.

NRC Request

2. *The staff notes that the GE-Hitachi Safety Analysis Report for CNS TPO, NEDC-33385P, Revision 0, November 2007, Section 6.7, "Fire Protection," states that the operator actions required to mitigate the consequences of a fire are not affected. Please verify that additional heat in the plant environment (from the MUR power uprate) will not interfere with required operator manual actions being performed at their designated time.*

NPPD Response

The operator manual actions that are being used for compliance with 10 CFR 50, Appendix R were reviewed. No operator manual actions have been identified in areas where environmental conditions, such as heat, would challenge the operator. Since this uprate is being performed at a constant pressure and temperature, the normal temperature environments are not affected by the MUR. Therefore, the MUR power uprate will have no impact on operator manual actions.

NRC Request

3. *The results of the Appendix R evaluation for the MUR power uprate are provided [in] Section 6.7, "Fire Protection," of the GE-Hitachi Safety Analysis Report for CNS TPO, NEDC-33385P, Revision 0, November 2007. However, this section does not discuss the time necessary for the repair of systems required to achieve and maintain cold shutdown nor the increase in decay heat generation following plant trips. Please verify that the plant can meet the 72-hour requirements in both 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L, with increased decay heat at MUR power uprate conditions.*

NPPD Response

A review was conducted of all repair activities that are credited to obtain and maintain cold shutdown. The CNS Appendix R analysis demonstrates that the station can reach cold shutdown with significant margin to the 72-hour requirements in 10 CFR 50 Appendix R, Sections III.G.1.b and III.L. No "time-critical" repairs would be required to reach or maintain cold shutdown. The MUR power uprate and the additional decay heat removal would not impact the ability to reach and maintain cold shutdown within 72 hours.

NRC Request

IV. The following question is provided from the Containment and Ventilation Branch (SCVB):

1. *On page 3 of Attachment 1 to your license amendment request (LAR) dated November 19, 2007 in Section 2.0, "PROPOSED CHANGE,"[ ] there is a bulleted item:*

> *ALLOWABLE VALUE on page 3.3-51 for TS Table 3.3.6.1-1, FUNCTION 1.c., Main Steam Line Flow - High, is revised from "≤ 142.7% rated steam flow."*

Given the information presented in Section 5.3.5, "Main Steam Line High Flow Isolation," on page 5-5 of the attachment NEDO-33385, Revision 0, "Safety Analysis Report for CNS TPO," to your LAR, it is not readily apparent how the proposed allowable value of "142.7%" was determined. Provide additional detail as to how the adjustment to the Main Steam Line High Flow Isolation Allowable Value was made to arrive at a value of "142.7%."

NPPD Response

The 142.7% was calculated using General Electric (GE) Setpoint Methodology. The values contained in NEDO-33385 are only estimates. NEDO-33385 estimates the rated steam flow at 2419 Megawatts Thermal (MWth) as "141.7% (better estimate 142.2%) of the TPO rated steam flow at 101.62% CLTP." However, CNS is committed to using GE Setpoint Methodology for these setpoints, thus the methodology must be followed.

The Analytical Limit for this setpoint is not changed. It is based on 150% of the originally rated steam flow. Steam flow is measured in terms of differential pressure across the flow element in the individual steam lines. A differential pressure of 121.24 pounds per square inch differential (psid) corresponds to 150% of rated steam flow at a rated power of 2381 MWth. However, the Analytical Limit has to be scaled to the new power level of 2419 MWth. Therefore, in the setpoint calculation for High Steam Flow, it is calculated that 150% flow at 2381 MWth = 147.64% flow at 2419 MWth. Both values of steam flow correspond to 121.24 psid.

From this Analytical Limit we establish an Allowable Value by following GE Setpoint Methodology (GE NEDC-31336P-A). The calculated Allowable Value is 111.73 psid, which corresponds to 142.7% flow at 2419 MWth.

NRC Request

V. The following questions are provided from the Reactor Systems Branch (SRXB):

1. Please identify and justify if any of the limiting analytical values that are part of the current Analysis of Record need to be modified as a result of operation at the MUR power uprate conditions at CNS.

NPPD Response

The Analytical Limits that were modified to support the CNS MUR are identified in Table 5-1 of Enclosure 3 of the LAR. The following is a summary of the changes identified in that table, with justification.

Parameter	Current	Thermal Power Optimization	Justification
APRM Flow Biased SCRAM			These changes to the Analytical Limits are based upon the methodology approved by the NRC in the GE Topical Report NEDC-32938P-A, Revision 2.
• TLO Flow Biased (%RTP)	0.66Wd + 74.8	0.75Wd + 65.6	
• SLO Flow Biased (%RTP)	0.66(Wd-ΔW) + 74.8	0.75(Wd-ΔW) + 65.6	
APRM Flow Biased Rod Block			
• TLO Flow Biased (%RTP)	0.66Wd + 64.0	0.75Wd + 54.8	
• SLO Flow Biased (%RTP)	0.66(Wd-ΔW) + 64.0	0.75(Wd-ΔW) + 54.8	
Turbine Stop Valve & Turbine Control Valve SCRAM & RPT Bypasses (%RTP)	30	29.5	All limits scaled for an uprate of 1.62% thermal. No change to Analytical Limit.
Main Steam Line High Flow Isolation % Rated Steam Flow (RSF) psid	150 % RSF 121.24 psid	147.6 % RSF 121.24 psid	
Rod Worth Minimizer LPSP (%RTP)	10 %RTP	9.85 %RTP	

NRC Request

2. *Describe the MUR power uprate core and batch size of GE fuel. If the MUR core is a mixed core, give details as required by the staff safety evaluation report for NEDC-32938P, "Generic Guidelines and Evaluations for GE BWR Thermal Power Optimization (TPO)" (Agencywide Documents Access and Management System Accession No. ML031050138). Discuss the impact of any new fuel type introduction on the proposed power uprate.*

NPPD Response

The MUR power uprate core will consist of 548 GE-14 type fuel assemblies. The batch size is 140 fresh GE-14 fuel assemblies. The MUR core will not be a mixed core. GE-14 fuel has been used at CNS since 2000. No new fuel types will be introduced in conjunction with the proposed power uprate.

NRC Request

3. *On page 5 of Attachment 1 of the submittal, it was stated that uncertainty in feedwater (FW) flow measurement is the most significant contributor to core power measurement uncertainty. In comparison, please discuss how significant is the boiling-water reactor recirculation flow measurement uncertainty, and justify how you assure that the uncertainty in recirculation flow measurement will not challenge the remaining uncertainty of 0.3 percent (2.0 percent - 1.7 percent) in core power measurement.*

NPPD Response

The core power measurement at CNS is done using a heat balance:

Core Thermal Power = Steam Energy + Reactor Water Cleanup Energy + Radiative Power Losses - Feedwater Energy - Control Rod Drive Energy - Recirculation Pump Energy

Recirculation flow is not a direct input to the heat balance, and any uncertainty in the measurement of recirculation flow will not challenge the remaining approximate 0.3% in the core thermal power uncertainty. Thus, the recirculation flow measurement uncertainty has no significance to core power measurement uncertainty.

NRC Request

4. *Please identify and discuss the significance of any differences between the actual CNS plant FW piping configuration and the model used at Alden Research Lab. The response should include deviations that can impact FW flow characteristics, such as pipe elbows, and any differences in geometry upstream of the FW flow instruments.*

NPPD Response

There is no difference between the CNS FW piping configuration and the model used at Alden Research Lab. The testing is described in Enclosure 5 of the LAR, and it included testing the spool piece in different axial configurations to address the uncertainty associated with field installation.

The leading edge flow meters (LEFM) will be installed in two straight sections of piping in the FW system. The installation location is downstream of reducers, which in turn are downstream of a flow split from a mixing tee. The installation location is upstream of 90 degree elbows. There is a possibility that the meters will need to be installed rotated to allow for transducer replacement. The reducers, flow split, mixing tee, and the elbows were included in the Alden Research Lab model for the CNS LEFMs. The meters were tested in both rotated and non-rotated configurations.

NRC Request

5. *If a leading edge flow monitor [meter] (LEFM) becomes inoperative, the staff understands that the existing flow nozzles, that have been calibrated with the last valid LEFM data, will be relied upon for a short period of time. If a defouling event should occur in the existing flow nozzles during this time period, an overpower condition could result. Please discuss this possibility.*

NPPD Response

A flow verification test and analysis on reactor feedwater was done in 1995 using an external ultrasonic meter. The conclusion was that the flow rate indicated by the ultrasonic meter was lower than the flow rate indicated by the existing flow nozzles by 0.007%. CNS inferred from these results that there was little or no fouling in the flow nozzles, which had not been tested since the start of plant operation in 1974. Based on these results, it is very unlikely that an overpower event would occur due to defouling during the 72-hour Allowed Outage Time requested.

NRC Request

6. *Please discuss the frequency of the listed preventive maintenance activities.*

NPPD Response

Preventive maintenance frequency is 18 months, based on vendor recommendations. These activities consist of physical inspections, power supply checks, back-up battery replacements, and internal oscillator frequency verification. These preventive maintenance activities are being implemented via the associated plant modification package.

NRC Request

7. *Please discuss the procedure for installation and testing of the flow measuring instrumentation, including the following additional information:*

   i) *An estimated time (hours or days) for installation and testing, and the potential radiation exposure to the technicians during installation and testing.*

   ii) *The Mode of plant operation (Modes 1 through 5) in which the plant is required (or preferred) to be for the purpose of installation and testing of the flow measuring instruments.*

   iii) *If the instruments (including electronics) are located in a radiation area of the plant, please discuss the impact of any expected radiation damage on the instrumentation and the resulting degradation of performance of the instruments.*

NPPD Response

The individual tasks with the estimated time for completion, the potential radiation exposure, and the allowed modes of plant operation are in the table follwing. This is based on estimates provided by the vendor based on their field experience.

Task	Estimated Time	Mode of Plant Operation
Install scaffolding to provide working access around LEFM spool pieces	2-3 days	Modes 1-3
Unpack and inspect LEFM spool pieces	1 hour	Modes 1-3
Route conduit from LEFM spool pieces to LEFM electronic cabinet	2-3 days	Modes 1-3 (on cabinet side) and 4-5 (once spool pieces installed)
Unpack and inspect LEFM electronics cabinet	1/2 day	Modes 1-3
Install LEFM electronics cabinet	1/2 day	Modes 1-3
Connect LEFM electronics cabinet to power feed	2 hours	Modes 1-3
Test and inspect transducer cables	1/2 day	Modes 1-3
Remove scaffolding around LEFM spool pieces	1-2 days	Modes 1-3
Commission system (set LEFM electronics cabinet software to site-specific parameters, set transducer gains, ensure meters operating correctly, and document meter configuration)	3-4 days	Mode 1 (at normal feedwater flow)
Removal of section of feedwater piping to accommodate LEFM spool pieces	1-2 days	Modes 4-5
Installation of LEFM spool pieces into feedwater line	1-2 days	Modes 4-5
Route transducer cables from the LEFM spool pieces to the LEFM electronics cabinet	1 day	Modes 4-5 (once spool pieces installed)
Connect field cables to LEFM electronics cabinet	1/2 day	Modes 4-5

Total dose anticipated for the installation and testing is estimated at 350-400 mR.

The instruments (including electronics) are designed to be stored and operated in a maximum gamma radiation field of 200 mR/hr (transducers) and 20 mR/hr (electronics cabinet). The transducers will be in the Turbine Building reactor feed pump room in an anticipated radiation field of 20 mR/hr at full power. The electronics cabinet will be in the corridor outside the Turbine Building reactor feed pump room in an anticipated radiation field of less than 1 mR/hr at full power. No radiation damage or degradation to the instruments (including electronics) due to such exposure is anticipated.

NRC Request

VI. The following questions are provided from the Technical Specifications Branch (ITSB):

1. *Explain how Surveillance Requirement (SR) 3.3.5.1, "Perform a Channel Check once per 12 hours," is sufficient to ensure that the necessary quality of Caldon CheckPlus systems and components are maintained and that facility operation will be within safety limits. This information is needed to ensure compliance with 10 CFR 50.36(d)(3).*

NPPD Response

Surveillance Requirement (SR) 3.3.5.1 is satisfied by the operator checking the maintenance status of the LEFM instrumentation at the cabinet in the Turbine Building. In addition to this local confirmation of status, the plant process computer will give a computer alarm message to the Control Room if the maintenance status of the LEFM instrumentation changes. The electronics cabinet has on-line, continuous monitoring of system parameters and the maintenance status of the cabinet will change if this monitoring reveals problems with the instrumentation.

NRC Request

2. *Correct the typo in Limiting Condition for Operation (LCO) 3.3.5, Required Action B.2.*

NPPD Response

CNS has changed "no greater than" to "not greater than" in the Technical Requirements Manual (TRM) Limiting Condition for Operation (LCO). Attachment 3 contains a final, for-information, copy of TRM Page 3.3-22.

NRC Request

3. *Identify the Attachment 2 Technical Specification markup pages that should contain a requirement to reset the Actions Condition, Applicability, Required Action, SR, Applicable Modes or other Specified Conditions, or LCO limits to the original licensing basis, if the Caldon CheckPlus System is inoperable for extended periods of time. This information is needed to ensure that (1) the lowest functional capabilities established in accordance with 10 CFR 50.36(d)(2)(i) are met, and (2) the SRs established in accordance with 10 CFR 50.36(d)(3) are met for the aforementioned time periods, which are not limited in duration from the time that Mode 1 is entered.*

NPPD Response

None of the Attachment 2 Technical Specification markup pages would need to contain requirements related to the LEFM being inoperable for an extended period of time. The Technical Specification requirement related to use of the CNS heat balance (which would be affected by LEFM inoperability for an extended period of time) is SR 3.3.1.1.2: "Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is less than or equal to 2% RTP plus any gain adjustment required by LCO 3.4.1, "Recirculation Loops Operating" while operating at greater than or equal to 25% RTP." The calculated power in the SR is taken from the heat balance. If the LEFM is inoperable for a period of greater than 72 hours, the calculated power will be limited to 2381 MWth per TRM TLCO 3.3.5, Condition B, and can still be used to adjust the APRMs.

Attachment 2

**Response to RAI,
Dated February 4, 2008,
Regarding LAR to Revise Technical Specifications for MUR Power Uprate
CNS, Docket No. 50-298, DPR-46**

The NRC RAIs are shown in italics and NPPD's response shown in block font.

NRC Request

I. The following questions are provided from the Electrical Engineering Branch:

1. Provide the existing and uprated power level in megawatts electric (MWe).

NPPD Response

Power Level	**Nominal Max. (Gross MWe)**	**Analyzed Max. (Gross MWe)**
Existing	815.0	828.97
Uprated	830.4	835.5

NRC Request

2. Provide a detailed comparison of existing ratings with uprated ratings and the effect of the power uprate on the following equipment:

a. main generator
b. normal station service transformer
c. startup station transformer
d. emergency station transformer

NPPD Response

a. Main Generator:

Power Level	Design		Max. Nominal		Admin Limits*	
	MVA @ 60 psig H2	MVA @ 50 psig H2	MWe @ 50 psig H2	MVAR @ 50 psig H2	MWe	MVAR
Existing	983	887.7	815.0	351.0	>815.0	150.0
Uprated	983	887.7	835.5	300.0	>835.5	150.0

* The Main Generator has an Administrative Limit of 150 Megavolt Ampere Reactive (MVAR) per CNS operating procedures. If it is necessary to produce more than 150 MVAR, the procedures direct that Operations enter the appropriate emergency procedure.

Operation at the uprated condition is not expected to have any effect on the operation of the Main Generator. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures.

b. Normal Station Service Transformer (NSST):

Power Level	Design MVA @ 0.85 pf	Design MWe @ 0.85 pf	Analyzed Max. Nominal MWe
Existing	33.6	28.56	24.0
Uprated	33.6	28.56	24.5

Operation at the uprated condition is not expected to have any effect on the operation of the NSST. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures.

c. Startup Station Service Transformer (SSST):

Power Level	Design MVA @ 0.85 pf	Design MWe @ 0.85 pf	Analyzed Max. Nominal MWe
Existing	30.0	25.5	24.0
Uprated	30.0	25.5	24.5

Operation at the uprated condition is not expected to have any effect on the operation of the SSST. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures.

d. Emergency Station Service Transformer (ESST):

Power Level	Design MVA @ 0.85 pf	Design MWe @ 0.85 pf	Analyzed Max. Nominal MVA @ 0.85 pf	Analyzed Max. Nominal MWe
Existing	12.6	10.71	10.5	8.925
Uprated	12.6	10.71	10.5	8.925

Note: The ESST supplies only the 1E busses. Therefore, the loading increase is negligible.

Operation at the uprated condition is not expected to have any effect on the operation of the ESST. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures.

NRC Request

3. *In Section 6.1.1 of Enclosure 3 of the license amendment request (LAR), the licensee states that the power factor (pf) for the generator is 0.85 whereas Table 6-1 of Enclosure 3 indicates a 0.58 pf. Please clarify the discrepancy.*

NPPD Response

The value for power factor of 0.58 in Table 6-1 was incorrect. The power factor for the Main Generator is 0.85. The 0.58 value is a typographical error. The Main Transformer's rating is 900/1008 Megavolt Ampere.

NRC Request

4. *In Section 6.1.1 of Enclosure 3 of the LAR, the licensee states that a grid stability study was performed and concludes that the proposed electrical output will not have any effect on grid stability or reliability. Provide details of the grid stability study and discuss in depth the assumptions, methodology, cases studied, and evidence to support the aforementioned conclusion.*

NPPD Response

A sensitivity analysis was performed to determine the impacts of different CNS generator output levels on the Loss of Coolant Accident (LOCA) analysis. The worst case with respect to grid stability is prior to a refueling outage (when power is decreasing due to plant coastdown) with winter peak conditions.

The results of the post-LOCA grid voltages, with CNS generation Net Output levels at 815 (Megawatts electrical) MWe (uprated) and 800 MWe (existing) prior to the trip, were evaluated. As expected, the pre-LOCA generation levels (815 MW versus 800 MW) had negligible impact on the post-LOCA voltage levels at the critical 161 kV, 69 kV and 4.16 kV buses at CNS. The 815 MWe and 800 MWe net generation levels modeled correspond to gross MWe generation levels of 839 MWe and 824 MWe, respectively.

The assumptions included a trip of the CNS generator concurrent with a Design Basis Accident LOCA. Voltage levels of the 345 kV, 161 kV and 69 kV transmission systems were calculated for the pre-accident and post-accident conditions, assuming seasonal loading variations, including summer and winter peaks. Individual transmission lines and generating stations were modeled as out of service for the seasonal loading variations, and the most limiting case was evaluated. As a result, it was determined that the impact of the CNS power uprate to the offsite power sources is negligible.

<u>NRC Request</u>

5. *For the power uprate of 1.62 percent, please identify the nature and quantity of megavolt ampere reactive (MVAR) support necessary to maintain post-trip loads and minimum voltage levels. Also, address how the power uprate would affect MVAR support.*

<u>NPPD Response</u>

The power uprate did not require a change to the nature or quantity of MVAR support for post-trip loads and minimum voltage levels. The ESST has a 5.4 MVAR capacitor bank that is used as required to maintain pre-accident voltage at or above 70.0 kV to assure that the second-level under-voltage relays do not actuate and cause load shedding during the sequential loading of the Emergency Core Cooling System (ECCS) loads. The SSST does not have an active MVAR support. However, Technical Specification LCO 3.8.1 is not met for the SSST if the Main Generator exceeds 150 MVARs out (an administrative limit established by CNS operating procedures). The 161 kV system voltage is maintained at or above a pre-accident voltage of 167.5 kV. The pre-accident voltage levels assure that the second-level under-voltage relays do not actuate and cause load shedding during the sequential loading of the ECCS loads. No changes were required to the existing MVAR support for either offsite sources as a result of the uprate.

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Attachment 3

TRM Page 3.3-22 For-Information
Appendix K MUR Power Uprate
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T 3.3 INSTRUMENTATION

T 3.3.5 Feedwater Flow Instrumentation

TLCO 3.3.5 Both Leading Edge Flow Meter CheckPlus instrumentation systems shall be OPERABLE.

APPLICABILITY: MODE 1 and THERMAL POWER > 2381 MWt

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more systems inoperable.	A.1 Verify no reduction in power $\geq$ 10% occurs during the 72 hour COMPLETION TIME of REQUIRED ACTION A.2.	Immediately
	<u>AND</u>	
	A.2 Restore required instruments to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of CONDITION A not met.	B.1 Initiate an orderly power reduction to $\leq$ 2381 MWt.	Immediately
	<u>OR</u>	
	B.2 Verify power is not greater than 2381 MWt.	Immediately

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©

Correspondence Number: NLS2008019

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
The completed response to NRC Question I.2 will be provided in a separate correspondence to the NRC.	NLS2008019-01	March 13, 2008