Nebraska Public Power District
Always there when you need us

NLS2008033
March 12, 2008

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

Subject: Response to Request for Additional Information for Question I.2 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 Stewart B. Minahan, Nebraska Public Power District, to the U.S. Nuclear Regulatory Commission, dated March 6, 2008, "Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications - Appendix K Measurement Uncertainty Recapture Power Uprate"

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 (RAI) Question I.2 sent on January 23, 2008 (Reference 1). The response to this question was not provided in the NPPD response, dated March 6, 2008 (Reference 2), due to the information not being available at that time. The attached information is in support of 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).

COOPER NUCLEAR STATION
P.O. Box 98 / Brownville, NE 68321-0098
Telephone: *(402) 825-3811 /* ***Fax:*** *(402) 825-5211*
www.nppd.com

A001
NRR

Attachment 1 contains a response to NRC RAI Question I.2 from Reference 1. To support readability, NPPD has restated NRC Question I.1 along with its response. This information is unchanged from that provided in Reference 2. The response to RAI Question I.2 represents new information. This attachment does not contain information considered proprietary as defined by 10 CFR 2.390. The information submitted by this letter (including attachment) 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 3-12-08

Sincerely,

Michael J. Colomb
Acting Vice President - Nuclear and
Chief Nuclear Officer

/dm

Attachment

cc: Regional Administrator w/ attachment
USNRC - Region IV

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

Senior Resident Inspector w/ attachment
USNRC - CNS

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

NPG Distribution w/o attachment

CNS Records w/ attachment

Attachment 1
Response to Request for Additional Information for Question I.2,
Dated January 23, 2008,
Regarding License Amendment Request to Revise Technical Specifications for
Measurement Uncertainty Recapture Power Uprate
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAI) Questions I.1 and I.2 are shown in italics. Nebraska Public Power District's (NPPD) response to each question is shown in block font. The response to RAI Question I.2 represents new information.

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 Cooper Nuclear Station (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

The Erosion/Corrosion Program at CNS uses the CHECWORKS SFA predictive code and actual wall thickness measurements to model wear rate caused by Flow Accelerated Corrosion (FAC). Using this program, the uprated conditions were input and compared with the current model. The uprated conditions in the model also include several plant modifications not related to the Measurement Uncertainty Recapture (MUR) (e.g., replacement of the low pressure turbines and the fourth and fifth stage FW heaters). It is impractical to separate the effect of the plant modifications from those of the MUR.

The wear rates calculated for this response were produced using a Pass 1 analysis with the CHECWORKS SFA predictive code. Pass 1 analysis does not include component inspection data like the Pass 2 analysis used to produce the response to Question I.1 previously supplied. The Pass 1 analysis is used to indicate the relative susceptibility of components within the piping to wear due to FAC, and the Pass 2 analysis is the method by which the Pass 1 results are calibrated so that the CHECWORKS predictions match observed conditions as closely as possible.

For determination of the effect on FAC due to a change in power, Pass 1 analyses were performed to compare the base case results to the uprated case. By performing the analyses using Pass 1, the models were not constrained by the line correction factor derived from previous inspection data. This is important since the past inspection results may not be indicative of future inspection results due to the change in power.

The average wear rate difference was calculated by determining the percent change in wear between the uprated Pass 1 and the base Pass 1 results. This percentage was multiplied with the base Pass 2 predicted wear results to determine the average wear rate difference. The uprate is predicted to result in an average wear rate increase of greater than two mils per year for:

- Extraction Steam piping to the third stage FW Heater.
- The drain piping from the fourth stage FW Heater.
- The drain piping from the fifth stage FW Heater.

None of the other piping systems are expected to have increased wear rates of greater than one mil per year (see Table 2).

The predicted increase in average wear in the Extraction Steam piping is due to the increase in wall velocity and the reduction of steam quality. The predicted increase in average wear for the drain piping is due to increased velocity for the fourth stage drains and approaching a more critical temperature for wear on the fifth stage drains. The MUR has no impact on oxygen concentration for the evaluated lines.

The estimated remaining service life for the three lines with average wear rates greater than two mils were reviewed to ensure that the expected increase in wear was acceptable for continued operation during the next fuel cycle (after the implementation of the power uprate).

Table 2: Appendix K Comparison*
(102% vs 100 %)

System	Average Wear Rate Difference (mils/yr)	Average Wear Rate Change (%)	Average Wall Velocity Difference (ft/s)	Average Wall Velocity Change (%)	Average Temp Difference (F)	Temp Change (%)	Average Quality Difference	Steam Quality Change (%)
BS: #2 Extraction[1]	-0.628	-3.50%	7.510	71.06%	-1.40	-0.67%	-0.036	-5.66%
BS: #3 Extraction	2.833	22.31%	3.066	13.36%	0.60	0.23%	-0.027	-3.27%
BS: #4 Extraction	Excluded - FAC resistant material							
BS: #5 Extraction	Excluded - FAC resistant material							
CH: Pumps to No. 1 Heaters	0.048	1.24%	0.312	2.24%	-0.10	-0.10%	0.000	0.00%
CH: #1 Heaters to #2 Heaters	-0.177	-4.94%	0.266	1.95%	-6.60	-3.91%	0.000	0.00%
CH: #2 Heaters to #3 Heaters	-0.091	-3.78%	0.279	1.64%	-6.30	-3.10%	0.000	0.00%
CH: #3 Heaters to #4 Heaters	-0.058	-2.66%	0.272	1.64%	-5.10	-1.97%	0.000	0.00%
CH: #4 Heaters to #5 Heaters	0.241	7.69%	0.308	1.77%	-2.10	-0.68%	0.000	0.00%
CH: #5 Heaters to Pumps	0.133	4.27%	0.261	1.68%	-3.00	-0.82%	0.000	0.00%
DR: No. 3 Heater Drains	-0.747	-13.69%	-0.614	-12.67%	-2.90	-1.37%	-0.001	0.00%
DR: No. 4 Heater Drains	2.091	12.88%	0.683	17.77%	-6.00	-2.23%	0.000	0.00%
DR: No. 5 Heater Drains[2]	4.663	16.70%	-2.025	-10.30%	-11.33	-3.54%	0.000	0.00%
DR: MSR Drains	0.019	0.77%	0.103	2.59%	1.00	0.27%	0.000	0.00%
MS: HP Turbine to MSR	0.117	0.98%	-1.780	-3.81%	5.50	1.47%	-0.003	0.00%
RF: Feedwater	0.239	4.40%	0.353	1.74%	-3.00	-0.82%	0.000	0.00%

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

1. The change in average velocity is not dominate due to the lower temperature of the BS-2 piping (206˚F).
2. The increase in wear is due to approaching a more critical temperature for wear.

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©

Correspondence Number: NLS2008033

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

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None	N/A	N/A