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NLS2012068
July 12, 2012

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

Subject: Response to Acceptance Review of Cooper Nuclear Station License Amendment Request to Adopt NFPA-805
Cooper Nuclear Station, Docket No. 50-298, DPR-46

Reference:
1. E-mail from Lynnea Wilkins, U.S. Nuclear Regulatory Commission, to Edward L. McCutchen, Nebraska Public Power District, dated June 21, 2012, "Acceptance Review of Cooper Nuclear Station LAR to Adopt NFPA-805 (ME8551)"
2. Letter from Brian J. O'Grady, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated April 24, 2012, "License Amendment Request to Revise the Fire Protection Licensing Basis to NFPA 805 Per 10 CFR 50.48(c)" (NLS2012006)

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District (NPPD) to respond to the Nuclear Regulatory Commission e-mail that provided LIC-109 acceptance review observations (Reference 1) on the NPPD NFPA 805 Transition License Amendment Request (Reference 2). The responses are provided in Attachment 1. Corresponding changes to the NFPA 805 License Amendment Request are provided in Attachment 2. NPPD has determined that the No Significant Hazards Consideration determination provided in Reference 2 remains bounding, and that this change therefore does not involve a significant hazard.

There are no commitments made in this submittal. Should you have any questions concerning this matter, please contact Todd Stevens, NFPA 805 Transition Project Manager, at (402) 825-5159.

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NRR

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

Executed on: 7/12/2012
(Date)

Sincerely,

Demetrius Willis

Demetrius L. Willis
General Manager of Plant Operations

DLW/wv

Attachment 1: Response to Acceptance Review of Cooper Nuclear Station License Amendment Request to Adopt NFPA-805

Attachment 2: Revisions to the License Amendment Request to Revise the Fire Protection Licensing Basis to NFPA 805 Per 10 CFR 50.48(c)

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 Acceptance Review of Cooper Nuclear Station
License Amendment Request to Adopt NFPA-805

The Nuclear Regulatory Commission (NRC) comments regarding the Acceptance Review of the Cooper Nuclear Station (CNS) License Amendment Request (LAR) to adopt NFPA 805 is shown in italics. The Nebraska Public Power District's (NPPD) response to each comment is shown in block font.

NRC Comment 1

LAR, Attachment G states that a majority of recovery actions (RA) currently credited have been assessed under the existing Fire Protection Program, which included field validation. However, according to LAR Attachment S, Implementation Item S-3.6, a confirmatory demonstration (field validation walk-through) of the feasibility for the credited NFPA 805 recovery actions will be performed and documented as part of LAR implementation. It appears that the extent to which the RA feasibility evaluation remains to be completed. Please provide the results of the completed evaluation and clarify the potential impact this remaining work may have on the results presented in the LAR.

NPPD Response

CNS Calculation NEDC 10-041 documents the results of the NFPA 805 RA feasibility assessment that evaluated the RA in Attachment G against the criteria outlined in Frequently Asked Question (FAQ) 07-0030, "Establishing Recovery Actions." Based on this assessment, the credited RA have been reviewed, and subsequently determined to be feasible with a high level of confidence. All RA, except those associated with manually opening the 4160VAC feeder breakers from the start-up transformer to Bus C and D, are part of the existing Appendix R program, and are documented in existing CNS procedures. These procedures have been field validated per the requirements of the CNS Procedure Change Process. The new RA to manually open the 4160VAC feeder breakers from the start-up transformer to Bus C and D involve removing control power fuses and tripping the breaker. These new actions are identical to existing actions to manually open other 4160VAC feeder breakers that exist in current CNS procedures. Therefore, it is concluded that there is a high level of confidence the new actions can be performed.

With the implementation of the Nuclear Safety Capability Assessment into the fire response procedures, there will be significant changes to the procedures due to the large reduction of required manual actions. The intent of Implementation Item S-3.6 is to do a confirmatory demonstration of the final procedures to assure all RA for a given fire area are in concert with each other, and to demonstrate coordination between operators performing the actions. The results of this confirmatory demonstration will be used as inputs for Implementing Item S-3.7.

<u>*NRC Comment 2*</u>

The additional risk of recovery actions is reported in LAR Table W-2 to be negative in several fire areas. This appears to be due to combining the risk reductions from plant modifications with the additional risk of recovery actions in the going forward plant, but not in the baseline (compliant) plant. To correct the calculation of the additional risk of recovery actions, plant modifications must be credited in the going forward plant as well as the baseline; in other words, the only difference in the going forward plant PRA and the baseline PRA is the credit for recovery actions. Please provide corrected results for the additional credit for recovery actions identified in Table W-2 on a fire area basis.

<u>NPPD Response</u>

Table W-2 has been revised, as provided in Attachment 2, to detail the remaining risk of RA on a fire area basis for the going forward plant probabilistic risk assessment (PRA). These risk values are positive numbers, and were calculated in conformance with FAQ 07-0030. There is no impact on results or conclusions due to the changes made to Table W-2.

<u>*NRC Comment 3*</u>

LAR section V.2 identifies deviations from the guidance in NUREG/CR-6850. For "Transient Fire Frequency," influence factors less than one were used. Also, a factor of 0.1 was used to modify both fire frequency from transients (bins 7, 25, and 37) and transient fires caused by welding and cutting (bins 6, 24, and 36) for certain fire zones. Please provide a simultaneous or composite sensitivity analysis of the impact on CDF, LERF, ΔCDF, and ΔLERF using NUREG/CR-6850 methods. It is expected that, concurrently, the credit for the factor of 0.1 which modifies the fire frequency would be removed, i.e. 1 would replace the 0.1, and that influence factors less than 1 would be replaced by those integers documented in Table 6-3 of NUREG/CR-6850.

<u>NPPD Response</u>

The transient fire frequency influencing factors utilized to generate the core damage frequency (CDF), large early release frequency (LERF), ΔCDF, and ΔLERF in the CNS LAR were based on proposed modifications/enhancements to the combustible and hot work controls for Fire Zone 8A (Auxiliary Relay Room) and Fire Zone 9A (Cable Spreading Room). The fire frequency was initially calculated using transient influencing factors identified in Table 6-3 of NUREG/CR-6850; however, it was determined that these two fire zones would receive enhanced administrative controls. NPPD identified that fractional values less than 1 were appropriate for the enhanced administrative controls instead of a value of zero (0) for the influencing factors to account for any unlikely violations of the new enhanced controls. The fractional values were utilized for only two of the influencing factors, therefore, the total transient influencing factor values for these two fire zones were greater than 1 (i.e., 1.15 and 3.15), which does not represent orders of magnitude differences in the ranges of influencing factors. The ignition frequency calculation weighting factors for other areas of the plant were analyzed based on the existing

occupancy, storage, and maintenance factors, and as such, it was not appropriate to increase the influencing factors for the other areas to account for the proposed enhanced controls in these two fire zones.

The sensitivity analysis was performed for the fire frequency results for transient fire scenarios in Control/Auxiliary/Reactor Building Plant Locations based on using only NUREG/CR-6850 methods by replacing the "Very Low" influencing factor values with maintenance and storage weighting factors identified in Table 6-3 of NUREG/CR-6850 for Fire Zone 9A (Cable Spreading Room) and Fire Zone 8A (Auxiliary Relay Room). Additionally, for three specific transient locations within fire zones (Fire Zone 3C and 3D, the area located above the TIP Room on the 903'-6" Elevation of the Reactor Building, and Fire Zone 2C, the floor areas immediately located around Instrument Racks 25-5 and 25-6 on the 931'-6" Elevation of the Reactor Building), the 0.1 factor was eliminated (changed to 1.0).

Since any adjustment in the transient influencing factors for a specific fire zone impacts all other fire zones in the Generic Plant Location - Control/Auxiliary/Reactor Building, fire zone transient frequencies also change by decreasing a very small amount. This very small decrease in other transient scenario frequencies was not included in sensitivity analysis, thus providing a worse case sensitivity result. There is no change to conditional core damage probability (CCDP) or conditional large early release frequency (CLERP).

The risk increases by 2.54E-06/year for CDF and 1.51E-06/year for LERF. The delta risk increases are 6.10E-08/year for CDF (difference between base delta CDF and sensitivity delta CDF) and 1.1E-08/year for LERF (difference between base delta LERF and sensitivity delta LERF).

The results are summarized as follows:

1. CDF increased by about 5%
2. LERF increased by about 12%
3. Delta CDF increased by less than 1%
4. Delta LERF increased by less than 1%

Conclusions did not change with respect to Regulatory Guide 1.174 acceptance guidelines. Further, there is considerable conservatism in the large early release frequency change for the following reason. Several scenarios in fire zones 8A and 9A result in the use of alternate shutdown. The Fire PRA model assumes that CLERP given core damage is 1.0. This conservatism more than compensates for the noted changes above.

NRC Comment 4

LAR Attachment U uses the term "vendor and utility" when describing the makeup of the team that performed the R.G. 1.200 Rev 1 peer review of the internal events PRA. This is confusing terminology potentially implying that the review was not an independent peer review as defined

in the ASME/ANS PRA Standard. Provide clarification that the peer review of the internal events PRA discussed in Attachment U met the requirements of Sections 1-6 and 2-3 of the ASME/ANS PRA Standard for a peer review. If not provide the F&Os and resolutions from the most recent full-scope peer review.

NPPD Response

The 2008 CNS internal events PRA peer review referenced in Attachment U of the LAR meets the requirements for an independent peer review as defined in the ASME/ANS PRA Standard. The 2008 CNS PRA peer review is a full-scope review of the Technical Elements of the internal events, at-power PRA. The peer review utilized and was conducted in accordance with the following standards and references:

- NEI 05-04, "Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard", Nuclear Energy Institute, Rev. 1, November 2007.
- ASME RA-Sc-2007, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications", August 2007.
- NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Rev. 1, January 2007.
- NRC memorandum, "Notice of Clarification to Rev. 1 of Regulatory Guide 1.200", FRN July 27, 2007, NRC ADAMS Accession number: ML071170054.

The peer review team was comprised of seven reviewers, of which, two were contractors and five were BWROG utility personnel. Reviewer qualifications and independence were confirmed and documented as part of the 2008 CNS peer review. The peer review team meets the current peer review team composition and personnel qualifications requirements of Sections 1-6 and 2-3 of the combined PRA standard (ASME/ANS RA-Sa-2009).

Attachment 2

Revisions to the License Amendment Request to Revise the Fire Protection Licensing Basis to NFPA 805 Per 10 CFR 50.48(c)

This attachment provides changes to the NFPA 805 License Amendment Request based on the responses to the Comments provided in Attachment 1, as well as for other clarifications. The changes are presented in underline/strikeout format.

1. Table W-2, CNS Fire Area Risk Summary, is revised as follows.

Table W-2 CNS Fire Area Risk Summary

Fire Area	Fire Area Description	NFPA 805 Basis	Fire Area CDF	Fire Area LERF	VFDR(s) (Yes/ No)	RA(s) (Yes/ No)	Fire Risk Eval Delta CDF	Fire Risk Eval Delta LERF	Additional Risk of RAs (CDF and LERF)
CB-A	RHR SW Booster Pump and Service Air Compressor Areas Emerg Condensate Storage Tank Area RPS Room 1A Seal Water Pump Area and Corridor	4.2.4.2	6.89E-07	7.03E-08	Yes	Yes	1.48E-07	4.62E-08	1.48E-07 4.62E-08
CB-A-1	Battery Room 1A DC Swgr Room 1A	4.2.4.2	3.43E-06	1.14E-07	Yes	Yes	9.49E-08	4.68E-08	9.49E-08 4.68E-08
CB-B	Battery Room 1B DC Swgr Room 1B	4.2.4.2	4.61E-06	1.85E-07	Yes	Yes	1.54E-07	6.94E-08	~~1.54~~4.34E-07 ~~6.94~~7.71E-08
CB-C	RPS Room 1B	4.2.4.2	1.74E-07	7.83E-09	Yes	~~Yes~~No	ε	ε	NA
CB-D	Computer Room Control Room and SAS Corridor Aux Relay Room Cable Spreading Room Cable Expansion Room	4.2.4.2	1.37E-05	4.20E-06	Yes	Yes	-1.53E-05	-1.44E-05	~~-1.53E-05~~4.74E-07 ~~-1.44E-05~~8.91E-08
DG-A	Div. 1 Diesel Generator	4.2.3.2	5.09E-06	9.18E-08	No	No	NA	NA	NA
DG-B	Div. 2 Diesel Generator	4.2.3.2	1.16E-06	9.00E-08	No	No	NA	NA	NA

Table W-2 CNS Fire Area Risk Summary

Fire Area	Fire Area Description	NFPA 805 Basis	Fire Area CDF	Fire Area LERF	VFDR(s) (Yes/ No)	RA(s) (Yes/ No)	Fire Risk Eval Delta CDF	Fire Risk Eval Delta LERF	Additional Risk of RAs (CDF and LERF)
IS-A	SW Pump Area Circ Water Pump and Traveling Screen Area	4.2.4.2	2.14E-07	2.61E-08	Yes	~~Yes~~No	2.36E-08	4.43E-10	~~2.36E-08~~ ~~4.43E-10~~NA
RB-A	RCIC and CS A Pump Room	4.2.4.2	2.52E-07	1.04E-08	Yes	Yes	7.90E-08	2.29E-09	7.90E-08 2.29E-09
RB-B	Core Spray B Pump Room Hydraulic Drive Pump Area	4.2.4.2	1.25E-07	1.15E-08	Yes	Yes	1.25E-07	1.15E-08	1.25E-07 1.15E-08
RB-CF	RHR Pump Rm 1A and 1C CRD Units-North 903' 6" South Corridor RHR HX-1A	4.2.4.2	1.67E-06	7.15E-08	Yes	Yes	7.90E-07	5.12E-08	~~7.90~~8.02E-07 ~~5.12~~5.22E-08
RB-DI	RHR Pump Room 1B and 1D HPCI Pump Room 903' 6" South Corridor CRD Units South RHR HX-1B	4.2.4.2	2.86E-06	5.05E-07	Yes	Yes	9.50E-07	2.00E-08	~~9.50E-07~~1.20E-06 ~~2.00E-08~~1.05E-07
RB-E	Suppression Pool Area	4.2.4.2	2.15E-07	5.14E-09	Yes	Yes	2.18E-09	2.44E-09	2.18E-09 2.44E-09
RB-FN	Rx Bldg 903' 6" NE Corner	4.2.4.2	1.84E-06	1.83E-08	Yes	Yes	-1.24E-07	-7.03E-10	~~-1.24~~1.60E-07 ~~-7.03E10~~5.12E-09
RB-J	SWGR Room 1F	4.2.4.2	1.25E-06	2.58E-07	Yes	Yes	1.43E-07	5.65E-08	1.43E-07 5.65E-08
RB-K	SWGR Room 1G	4.2.4.2	1.64E-06	2.64E-07	Yes	Yes	-2.08E-06	-2.15E-06	~~-2.08E-06~~6.09E-07 ~~-2.15E-06~~1.09E-07

Table W-2 CNS Fire Area Risk Summary

Fire Area	Fire Area Description	NFPA 805 Basis	Fire Area CDF	Fire Area LERF	VFDR(s) (Yes/ No)	RA(s) (Yes/ No)	Fire Risk Eval Delta CDF	Fire Risk Eval Delta LERF	Additional Risk of RAs (CDF and LERF)
RB-M	RWCU Recirc Pumps and Corridor	4.2.4.2	4.46E-07	2.21E-08	Yes	Yes	-6.32E-07	-2.76E-09	~~-6.32E-07~~7.41E-08 ~~-2.76E-09~~1.60E-08
RB-N	RHR HX-1B Regenerative HX Areas RWCU Recirc Pumps and Corridor	4.2.4.2	2.63E-07	2.48E-08	Yes	~~Yes~~No	6.43E-08	5.70E-10	~~6.43E-08 5.70E-10~~NA
RB-P	RB Elevator and accessway Area RB HVAC Areas Fuel Pool, HX, CRD Repair Room, and Raw Water Cleanup Areas Reactor MG Set Oil Pump Area	4.2.4.2	9.02E-08	1.97E-08	Yes	Yes	4.63E-08	1.72E-08	4.63E-08 1.72E-08
RB-T	SBLC Pump Tank and Accessway (Zone 5A) and Refueling Floor (Zone 6)	4.2.3.2	3.80E-08	6.53E-09	No	No	NA	NA	NA
RB-V	Rx MG Set Area	4.2.4.2	3.15E-08	7.19E-09	Yes	Yes	1.88E-08	6.31E-09	1.88E-08 6.31E-09
TB-A	Turbine Building	4.2.4.2	9.56E-06	3.87E-06	Yes	Yes	6.77E-06	3.33E-06	6.77E-06 3.33E-06
TB-C	Steam Tunnel	4.2.4.2	1.09E-08	1.23E-09	Yes	~~Yes~~No	ε	ε	NA
Yard	Yard Outside of Buildings	4.2.3.2	1.35E-06	6.30E-07	No	No	NA	NA	NA
	Total		**5.07E-05**	**1.05E-05**			**-8.71E-06**	**-1.29E-05**	**~~-8.71E-06~~1.12E-05 ~~-1.29E-05~~3.97E-06**

Reference: Response to Comment 2 from Attachment 1. Additionally, fire areas CB-C, IS-A, RB-N, and TB-C contain corrections. No Recovery Actions (RA) are credited with the VFDR for these fire areas; thus, column "RA(s) (Yes/No)" is revised to "No," and the associated additional risk of RA for these fire areas is not applicable.