Nebraska Public Power District
Always there when you need us

NLS2015068
June 22, 2015

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Nebraska Public Power District's Response to Request for Additional Information Associated with Near-Term Task Force Recommendation 2.1, Seismic Hazard and Screening Report
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:
1. NRC letter, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012

2. NPPD letter to NRC, "Revision to Nebraska Public Power District's Response to Nuclear Regulatory Commission Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident" dated February 11, 2015

3. NRC Letter to NPPD, "Cooper Nuclear Station - Request for Additional Information Associated with Near-Term Task Force Recommendation 2.1, Seismic Hazard and Screening Report," dated May 4, 2015

Dear Sir or Madam:

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued Reference 1 to all power reactor licensees and holders of construction permits in active or deferred status. By Reference 2, Nebraska Public Power District (NPPD), provided the revised Seismic Hazard Evaluation and Screening Report for Cooper Nuclear Station. In Reference 3, the NRC issued a request for additional information (RAI). The purpose of this letter is for NPPD to respond to Reference 3. The specific responses to the RAI questions are provided in the attachment.

There are no new regulatory commitments contained in this letter.

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

A010
NRR

Should you have any questions or require additional information, please contact Jim Shaw, Licensing Manager, at (402) 825-2788.

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

Executed On 6/22/15
(Date)

Sincerely,

Kenneth Higginbotham
General Manager of Plant Operation

/bk

Attachment: Response to Nuclear Regulatory Commission Request for Information on the Revised Seismic Hazard Evaluation and Screening Report for Cooper Nuclear Station

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

Director, w/ attachment
USNRC - Office of Nuclear Reactor Regulation

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

Senior Resident Inspector, w/ attachment
USNRC - CNS

CNS Records, w/ attachment

NPG Distribution, w/o attachment

Attachment

Response to Nuclear Regulatory Commission Request for Information on the Revised Seismic Hazard Evaluation and Screening Report for Cooper Nuclear Station

Cooper Nuclear Station, Docket No. 50-298, DPR-46

By letter dated February 11, 2015, Nebraska Public Power District (NPPD) submitted a revised Seismic Hazard Evaluation and Screening Report for Cooper Nuclear Station (CNS). In a letter dated May 4, 2015, the Nuclear Regulatory Commission (NRC) transmitted the following request for additional information (RAI):

Background:

After reviewing IPEEE Adequacy Evaluation and IPEEE High Confidence of a Low Probability of Failure (HCLPF) spectra (IHS) Development part of the Seismic Hazard and Screening Report and the earlier IPEEE submittal for CNS, the NRC staff has four requests for additional information. The information will be reviewed by the NRC staff as part of the NRC staff evaluation of the IPEEE adequacy and, if needed, to support a follow-on audit of the items addressed in the requests.

NRC RAI 1:

Section 3.2 of the Seismic Hazard and Screening Report (SHSR) states that the control point elevation of 869.5 ft. was used for the comparison of the GMRS, IHS and SSE. Section B.4.1 of Appendix B of the SHSR, states that the control point of the Review Level review Earthquake motion applied in SHAKE, as part of the IPEEE adequacy demonstration, was 902 feet. Please describe the impact, if any, this apparent control point discrepancy has in the final screening results.

NPPD Response:

As noted, the Review Level Earthquake (RLE) was developed using a control point at the free surface elevation of 902 feet. If the RLE were to be determined based on the Ground Motion Response Spectra (GMRS) control point elevation of 869.5 feet the filtering effect of the structural fill would be removed and the RLE spectral response peak would be lowered and shifted towards higher frequencies.

The minimum margin in the 1 to 10 Hz range between the IPEEE High Confidence of a Low Probability of Failure (HCLPF) spectra (IHS) and GMRS is approximately 20% near 10 Hz with additional margin of up to 400% at frequencies near 1 Hz[1-1]. The existing margin between the IHS and GMRS at low frequencies is considered sufficient to obviate the need to perform the risk evaluation screening (1 to 10 Hz) due to these differences. CNS currently screens in for the high frequency exceedance (10+ Hz) and plans to complete the high frequency evaluation.

The results of the Electric Power Research Institute (EPRI) High Frequency Program[1-2] have shown that there is not an increase in relay sensitivity at high frequencies:

> The primary result of the High Frequency Program is the general conclusion that while some components are sensitive to vibration, there was no unique high frequency sensitivity identified. Some components are sensitive to vibration in general, but that sensitivity is not increased in the high-frequency region.

Based on the amount of margin between the IHS and the GMRS in the 1 to 10 Hz range as well as the plan to complete the high frequency evaluation there is no impact to the final screening results due to the difference in the elevation of the GMRS control point and the IHS.

References:

1-1. NEDC 15-065, SHSR RAI Responses, Rev. 0

1-2. EPRI 3002002997, High Frequency Program: High Frequency Testing Summary, September 2014

<u>NRC RAI 2:</u>

Section C.2.2 of Appendix C of the SHSR discusses the enhanced liquefaction assessment performed for CNS. The analyses performed by the licensee show factors of safety against liquefaction generally on the order of 1.4 or greater over a significant portion of the structural fill profile (Figures C.2-1 and C.2-2). Although significant loss of shear strength is not expected when the factor of safety is 1.4 or greater, some seismic induced settlement is anticipated. Please specify how much seismic induced settlement/differential settlement is expected and what implications does this amount of settlement have on the success of the safe shutdown path?

<u>NPPD Response:</u>

Five soil borings were evaluated using the method of Tokimatsu and Seed[2-1] to estimate the potential seismic induced settlement and differential settlement within the structural fill. The method of Tokimatsu and Seed[2-1] is identified in NUREG/CR-5741[2-2] as "...appropriate for hazard screening purposes and is summarized below for estimation of settlement due to liquefaction of granular soils." The method of Tokimatsu and Seed[2-1] estimates the seismic induced settlement using the dependence of volumetric strains from dissipation of excess pore water pressures on the cyclic stress ratio (CSR) and the initial relative density of the soil, which is quantified by the Standard Penetration Test N value, $(N_1)_{60}$, with adjustments for fines content effects. The CSR values are the same as those in the Seismic Hazard and Screening Report (SHSR) and are based on the IHS.

The same soil parameters (i.e., total unit weight, fines content, etc.) were used for the liquefaction potential evaluation and the seismic induced settlement evaluation.

From the resulting analysis of seismic induced settlement in each of the five soil borings from the structural fill, the average total settlement is estimated to be 0.02 and 0.03 inches for the observed and flood groundwater conditions, respectively. The maximum differential settlement is estimated to be 0.06 and 0.08 inches for the observed and flood groundwater conditions, respectively. The maximum differential settlement is estimated as the difference between the minimum and maximum seismic settlement value in the five soil borings for samples of structural fill below the foundation levels[2-3].

The CNS Preliminary Safety Evaluation Report (PSAR)[2-4] stated that allowable differential settlements on the order of an inch were generally considered acceptable for no impact on performance of the structure. These results confirm the estimated seismic induced settlement in the structural fill is less than one inch for the IHS. Therefore, seismic induced settlement/differential settlement will not affect the success of the safe shutdown path.

References:

2-1. Tokimatsu, K. and H.B. Seed, Simplified Procedures for the Evaluation of Settlements in Clean Sands, Earthquake Engineering Research Center, Report No. UCB/EERC-84/16, October 1984

2-2. NUREG/CR-5741, Technical Bases for Regulatory Guide for Soil Liquefaction, October 1999

2-3. NEDC 15-065, SHSR RAI Responses, Rev. 0

2-4. Consumer Public Power District, CNS PSAR, Volume III, Appendix A – Geology, Seismology, Subsurface Investigation, and Foundation Recommendations, 1967

<u>NRC RAI 3:</u>

Section C.2.3 of Appendix C of the SHSR discusses the enhanced liquefaction assessment for the native alluvium underlying the Radwaste Building. CNS's SHSR states that the factor of safety against liquefaction is typically less than 1.0 in the native alluvium below the Radwaste Building. Liquefaction may cause significant strength loss leading to bearing capacity failure and large differential settlements resulting in significant structural damage to this building and its foundation. In addition, the licensee states that in the event of liquefaction, the Radwaste building will rotate away from the Control and Reactor buildings. Please provide the following:

a) *Additional information supporting the licensee's conclusion that the Radwaste Building can only undergo rotation away from the Control and Reactor buildings. The information should include details about the estimated seismically induced differential settlement, the structural foundation, and the structure itself that supports the aforementioned conclusion.*

b) *In the case of a liquefaction event that results in the complete overturn or collapse of the Radwaste building, please provide reassurance that a total loss of this building doesn't affect the licensee's coping capabilities.*

NPPD Response:

3(a) The SHSR shows the area where the Radwaste Building (RWB) is not fully supported by structural fill in Figure C1-1 in low detail. The thickness of the structural fill below the RWB foundation increases from plant northwest to plant southeast as shown in Figure 1 below. The east and southeast of the RWB is fully supported by structural fill. The potential settlement of the northwest end of the RWB is in the range of 3 to 14 inches with an average of 8.7 inches[3-1]. As noted in the Response to RAI 2 above, the maximum settlement below the portion of the RWB fully supported by structural fill is 0.02 inches under normal conditions and 0.03 inches if the IHS occurs during flood conditions.

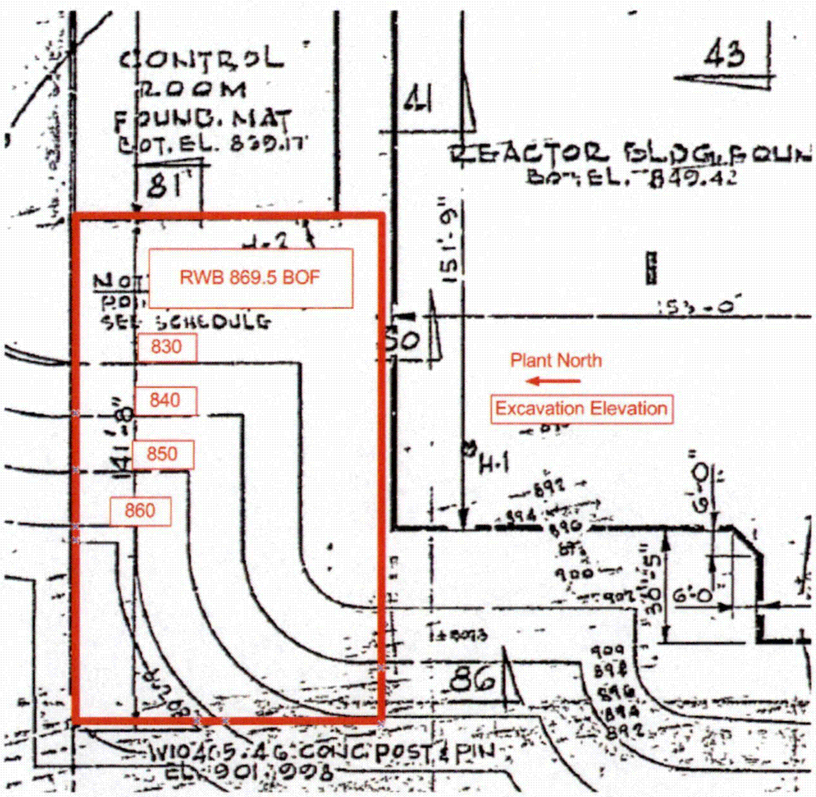

Figure 1 - Excavation Elevations Below RWB

The increase in settlement beneath the northwest (plant) corner provides the basis for the statement that the RWB will only rotate away from the Control Building and Reactor Building. Since the east and southeast will experience less settlement than the northwest end of the RWB, the differential settlement would result in rotation away from the Control Building and Reactor Building.

3(b) Section C2.3 of Appendix C of the SHSR states "As indicated in the CNS USAR, the below-grade portion of the Radwaste Building is a Class I structure not required for safe shutdown, and the above-grade portion of the Radwaste Building is a Class II structure."

The definition of a Class I and Class II structure classifications from the CNS Updated Safety Analysis Report (USAR)[3-2] is:

> Class I - This class includes those structures, equipment, and components whose failure or malfunction might cause or increase the severity of an accident which would endanger the public health and safety. This category includes those structures, equipment, and components required for safe shutdown and isolation of the reactor.
>
> Class II - This class includes those structures, equipment, and components which are important to reactor operation, but are not essential for preventing an accident which would endanger the public health and safety, and are not required for the mitigation of the consequences of these accidents. A Class II designated item shall not degrade the integrity of any item designated Class I.

The RWB below-grade portion was designed to withstand Seismic Class I earthquake to preclude leakage of radioactive liquid out of the structure. Failure of the structure to contain leakage could have a potential impact on public health and safety and this is the basis for the designation as Class I. The systems within the RWB structure are not safety-related[3-3] and therefore any piping systems which penetrate through the structure walls, which could be impacted by the building rotation, are not safety-related.

As noted in the response to 3(a) above the rotation of the RWB will be to the west / northwest based on the different amounts of seismic settlement in the native alluvium and structural fill. The structures and components adjacent to the RWB which could be impacted by building rotation to the west include the Multi-Purpose Facility (MPF), Augmented Radwaste Building (ARB), and Administration Building. None of these four buildings, or the components housed within them, are relied upon to support the CNS safe shutdown [3-2]. The MPF and ARB are designated as Class II structures, similar to the above grade portion of the RWB, and therefore not required for safe shutdown.

Based on the above, a total loss of the RWB will not impact the coping capabilities at CNS.

References:

3-1. NEDC 14-022, Soil Failure Evaluation, Revision 1

3-2. NPPD, CNS USAR Section XII-2.1, Classification of Structures and Equipment

3-3. Letter, Burns & Roe to NPPD, "NRC IE Bulletin 79-14, Piping Systems in the Radwaste Building," August 7, 1979

NRC RAI 4:

As part of the IPEEE screening process described in the SPID, licensees are required to provide background information in order to demonstrate the adequacy of the results. Specifically, licensees are required to "Confirm that any identified deficiencies or weaknesses to NUREG-1407 in the plant specific NRC Safety Evaluation Report (SER) are properly justified to ensure that the IPEEE conclusions remain valid." According to the IPEEE SER, the staff noted that only low pressure injection systems were selected for success paths and included as part of the safe shutdown equipment list (SSEL) process. As such, high pressure injection systems, including Reactor Core Isolation Cooling and High Pressure Coolant Injection were not included as part of the success path evaluation and the SSEL selection process. The NRC SER states that not including a high pressure injection system differs from paths suggested in EPRI NP-6041, and other previous IPEEE submittals, as a "first line of defense" that responds automatically. The exclusion makes the demand on low pressure systems more significant, which in turn, makes the demand on operator actions more significant since the automatic initiation circuitry for depressurization and initiation of the low pressure systems was also not included in the SSEL. The staff has determined that the following critical information was not provided and was needed to demonstrate the accuracy of the IPEEE results

a) Explanation on how the reliance on operator/manual actions were taken into account in the analysis to ensure an operator error could not fail a success path.

b) Justification of the completeness and diversity of the SSEL list.

NPPD Response:

4(a) The reliance on operator/manual actions was taken into account in the IPEEE analysis[4.1] as described in the SHSR, Appendix B, Section B.4.7, "System Modeling":

> In the CNS IPEEE report, the non-seismic failures and human actions were evaluated using the CNS Probabilistic Safety Assessment (PSA) model. The turbine trip combined with a loss of off-site power was modeled in the CNS PSA using only the SSEL systems to predict the plant post-seismic reliability. The non-seismic-related conditional plant response was estimated to be 7.17E-03/day using the modeled event sequence. As stated in the CNS IPEEE report, the post-event reliability provides assurance that the overall post-seismic reliability of CNS is high when also considering the low recurrence frequency of the RLE; therefore, the event-related core damage frequency (CDF) is low.
>
> Since the quantified plant response included some credited human actions, CNS performed an additional assessment where the conditional plant response was modified by overstating the human probability factors by two orders of magnitude. The results of the assessment showed that the conditional response was increased by only 2% over the 7.17E-03/day value previously indicated. Therefore, CNS concluded that the post-seismic event human actions are not significant.

The IPEEE Safety Evaluation Report (SER) [4-3] noted that manual depressurization of the reactor is required for both success paths, and the effects of operator failure to depressurize needed to be discussed. In an RAI response on this topic[4-4], CNS clarified that the Automatic Depressurization System would act automatically if the high-pressure systems were to malfunction during a seismic event:

> It would be incorrect to assume that manual depressurization of the reactor vessel is required in order to enable the low pressure injection systems to function. Comparing high pressure injection with low pressure injection for the success path, both paths function in an automatic fashion, if left to themselves. If high pressure injection were to malfunction when required to operate during a seismic event, the automatic depressurization system (ADS) would act automatically to depressurize the reactor vessel, in order to accommodate low pressure injection systems. The fact that the control room operators have the option of inhibiting ADS action, for the case that they might prefer manual control of reactor pressure, does not affect existing automatic system features.

4(b) The exclusion of the high pressure systems (High Pressure Coolant Injection and Reactor Core Isolation Cooling) from the safe shutdown equipment list (SSEL) was addressed during the IPEEE RAI process[4-2][4-3][4-6] as described in the SHSR submittal, Appendix B, Section B.4.3:

> During the BNL review of the CNS IPEEE report, NRC developed an RAI regarding the use of low-pressure injection systems only and requested the basis for not including the high-pressure systems High-Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) in the SSEL. As noted in Section 2.4 of the BNL TER included in the SER in NRC letter, the RAI response clarifies the rationale for selecting the low pressure injection system for the success paths. The RAI response states that based on conservative reasoning it may be operationally desirable to depressurize during a postulated RLE; therefore, the Automatic Depressurization System (ADS) was included in both success paths. The BNL TER included in the SER in NRC letter indicates that the response to the RAI is reasonable and that the equipment in the SSEL can provide two success paths for safe shutdown under transient and small SBLOCA conditions.

Automatic initiation circuitry for depressurization and initiation of the low pressure systems was reviewed as part of the A-46 analysis as described in Response to RAI[4-4] question 9(d):

> The A-46 systems analysis does not credit any automatic initiation to align the plant systems for safe shutdown. Although circuitry which provides for automatic initiation of A-46 systems was analyzed during the relay review process, automatic initiation (where favorable to safe shutdown) is not assumed to occur.

The review of circuitry for A-46 systems, specifically relays, was discussed in the IPEEE Report Section 3.1.2.1.9[4-1]:

> The relay screening techniques utilized were similar to those utilized for the more strict USI A-46 relay review. For each safe shutdown component identified as requiring the evaluation type "Relay", initial relay screening was performed to identify the set of relays which supported the associated SSEL component. Any device with electrical contacts was considered a relay for purposes of the initial screening. These relays were classified as non-essential or essential by considering the effects of contact chatter upon the required safe shutdown function of the associated SSEL component.
>
> Non-essential relays are those relays for which contact chatter could not adversely impact the safe shutdown function of the associated SSEL component(s). Non-essential relays include devices such as mechanically actuated control and limit switches which are considered to be seismically rugged and, therefore, not vulnerable to contact chatter.
>
> Essential relays are those relays for which contact chatter could adversely impact the safe shutdown function of the associated SSEL component(s). Manufacturer and model number data for the subset of relays classified as essential was reviewed to determine if any- were "low ruggedness" models as identified in Appendix E of EPRI NP-7148-SL, "Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality". No "low ruggedness" relays were identified as a result of this effort.

The IPEEE SER[4-3] determined that the SSEL inclusion of only low pressure systems was reasonable and that the equipment in the SSEL can provide two success paths for safe shutdown under transient and small SBLOCA conditions.

The IPEEE SSEL did not include the automatic initiation circuitry for depressurization and initiation of the low pressure systems because these were not credited pathways. The automatic initiation circuitry was evaluated as part of the USI A-46 SSEL and found to be acceptable. As noted in the IPEEE report[4-1] the evaluations for USI A-46 were more strict than required for IPEEE and no "low ruggedness" relays were identified.

References:

4-1. Nebraska Public Power District (G. R. Horn), Letter NLS960143 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Individual Plant Examination for External Events (IPEEE) Report - 10 CFR 50.54(f)," October 30, 1996

4-2. U.S. Nuclear Regulatory Commission (James R. Hall), Letter to Nebraska Public Power District (G. R. Horn), "Request for Additional Information Related to the Individual Plant Examination of External Events (IPEEE) for the Cooper Nuclear Station (TAC No. M83611)," June 3, 1998

4-3. U.S. Nuclear Regulatory Commission (Mohan C. Thadani), "Letter to Nebraska Public Power District (John H. Swailes), Cooper Nuclear Station – Review of Individual Plant Examination of External Events (TAC NO. 83611)," April 27, 2001

4-4. Nebraska Public Power District (John H. Swailes), Letter NLS980192 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Response to Request for Additional Information related to USI A-46 and Notification of Outlier Resolution," January 25, 1999

4-5. Electric Power Research Institute, Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, 1025287, EPRI, Palo Alto, California, February 2013

4-6. Nebraska Public Power District (John H. Swailes), Letter NLS990008 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Response to Request for Additional Information – Individual Plant Examination for External Events (IPEEE)," January 28, 1999