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NLS2015064
June 9, 2015

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

Subject: Response to Nuclear Regulatory Commission Requests for Additional Information for License Amendment Request to Add Residual Heat Removal System Containment Spray Function and a Supplement to the License Amendment Request
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:
1. Email from Siva Lingam, U.S. Nuclear Regulatory Commission, to Jim Shaw, Nebraska Public Power District, dated May 20, 2015, "Cooper Nuclear Station - Requests for Additional Information (RAIs) for the License Amendment Application to Revise Technical Specifications to Add Residual Heat Removal System Containment Spray Function (TAC No. MF5584)"
2. Letter from Oscar A. Limpias, Nebraska Public Power District, to the U.S. Nuclear Regulatory Commission, dated January 15, 2015, "License Amendment Request to Revise Technical Specifications to Add Residual Heat Removal System Containment Spray Function"

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District to respond to the Nuclear Regulatory Commission's Requests for Additional Information (RAIs) (Reference 1) related to the Cooper Nuclear Station License Amendment Request to revise Technical Specifications to add Residual Heat Removal System Containment Spray function (Reference 2). In addition, a supplement to the license amendment request is necessary because of revisions performed to a calculation related to containment spray flowrates.

The responses to the specific RAI questions are provided in Attachment 1. The supplement to the license amendment request is described in Attachment 2.

The supplement to the license amendment request neither impacts the conclusions of the no significant hazards consideration evaluation that was performed pursuant to 10 CFR 50.91(a)(1), nor the environmental consideration performed pursuant to 10 CFR 51.22.

A001
NRR

This letter does not contain any new regulatory commitments.

If you have any questions concerning this matter, 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: 6/09/15
(Date)

Sincerely,

Oscar A. Limpias
Vice President - Nuclear
and Chief Nuclear Officer

/dv

Attachments:
1. Response to Nuclear Regulatory Commission Requests for Additional Information (RAI) for License Amendment Application to Revise Technical Specifications to Add Residual Heat Removal (RHR) System Containment Spray Function (TAC No. MF5584)
2. Supplement to License Amendment Request to Revise Technical Specifications to Add Residual Heat Removal (RHR) Containment Spray Flowrates

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

NPG Distribution w/o attachment

CNS Records w/ attachment

Attachment 1

Response to Nuclear Regulatory Commission Requests for Additional Information (RAI) for License Amendment Application to Revise Technical Specifications to Add Residual Heat Removal (RHR) System Containment Spray Function (TAC No. MF5584)

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

The Nuclear Regulatory Commission (NRC) request for additional information regarding the request to add RHR System Containment Spray Function to the Technical Specifications is shown in italics. The Nebraska Public Power District (NPPD) response to the request is shown in normal font.

The first paragraph on page 6 of Attachment 1 of the application dated January 15, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15021A127, [as supplemented by letter dated May 4, 2015 (ADAMS Accession No. ML15132A652)], states:

> *"The analyses demonstrate that the temperature and pressure reduction capacity of the RHR [Residual Heat Removal] Containment Spray System is adequate to maintain the primary containment conditions within design limits. The RHR Containment Spray system satisfies Criterion 3 of [Title 10 of the Code of Federal regulations (10 CFR), Part 50, Section] 10 CFR 50.36(c)(2)(ii)."*

Also, in the third paragraph on page 6 of Attachment 1 of the application dated January 15, 2015, it was stated:

> *"A plant specific realistic model was developed to determine drywell airspace temperature response."*

<u>*RAI-1*</u>

Was the plant specific realistic model mentioned above reviewed and approved by the NRC? If not, then justify acceptability of the model for the design-basis small steam line break (SSLB) accident.

<u>NPPD Response:</u>

The plant specific realistic model developed for the SSLB analysis has not been reviewed or approved by the NRC. The SSLB analysis utilized General Electric's (GE) SHEX-04V computer code for the first 100,000 seconds of the long term containment response. The use of the SHEX-04V code has been accepted by the NRC for calculating the response of the containment during an accident or transient event and has been applied to the evaluation of containment response for many Boiling Water Reactor (BWR) plants. This code is currently utilized in the long term containment responses documented in the Cooper Nuclear Station (CNS) Updated Safety

Analysis Report (USAR) Chapters VI and XIV. NRC letter from A. Thadani to G.L. Sozzi (GE), July 13, 1993, "Use of SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," summarizes the licensee expectation when utilizing the SHEX computer code. A bench mark of the computer code to an NRC reviewed analysis is required. A bench mark analysis was completed and documented in GE report GE-NE-T23-00786-00-01, "Cooper Nuclear Station Containment Analysis Project - NPSH [Net Positive Suction Head]." This bench mark analysis showed that SHEX-04V yielded similar or conservative results compared to the original Case C Final Safety Analysis Report (FSAR) analysis.

RAI-2

Describe the design-basis loss-of-coolant Accident (LOCA) for CNS.

NPPD Response:

The CNS design-basis accident (DBA) LOCA is defined as a complete circumferential break of one of the Reactor Recirculation loop lines.

In the short-term DBA-LOCA, the reactor will automatically scram due to high drywell pressure. Main steam line isolation will occur due to low reactor water level. No mechanical Safety Relief Valve actuation will occur because of the rapid reactor vessel depressurization and large rate of reactor fluid and energy inventory loss through the break. Shortly after the postulated pipe break, the Emergency Core Cooling System (ECCS) automatically begins to pump water from the plant emergency condensate storage tank and/or the suppression pool into the reactor pressure vessel to flood the reactor core. Following vessel flooding and drywell/wetwell airspace pressure equalization, suppression pool water is continually recirculated from the pool to the reactor vessel by the ECCS pumps. The suppression pool cooling mode of the RHR system is manually actuated to remove energy from the suppression pool to return the Primary Containment to normal temperature conditions.

For the long-term DBA-LOCA, the Core Spray system removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal-water reaction to less than 0.1 percent. The Core Spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water flows into the suppression chamber via the drywell-to-suppression chamber vent pipes. Steam flow is negligible. The energy transported to the suppression chamber water is then removed from the Primary Containment by the RHR heat exchangers.

Prior to activation of the RHR containment cooling mode (arbitrarily assumed at 600 seconds after the accident), the RHR pumps, in the Low Pressure Coolant Injection (LPCI) mode, have been adding liquid to the reactor vessel. After the reactor vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the recirculation line break into the drywell. This flow, in addition to heat losses to the drywell walls, offers considerable cooling to the drywell and causes a depressurization of the Primary Containment as the steam in the drywell is

condensed. At 600 seconds, the RHR pumps are assumed to be switched from the LPCI mode to the RHR containment cooling mode.

To assess the Primary Containment long term response after the accident, an analysis was made of the effects of various containment spray and cooling combinations and parameter relaxations. For all cases, one of the Core Spray loops is assumed to be in operation. The long-term pressure and temperature response of the Primary Containment to the DBA-LOCA was analyzed for the following RHR containment cooling mode conditions. The limiting cases are Case E and F.

- Case A - Operation of both RHR cooling loops - 4 RHR pumps, 4 RHR Service Water Booster pumps, 3 Service Water pumps, and 2 RHR heat exchangers - with containment spray.

- Case B - Operation of one RHR cooling loop with 2 RHR pumps, 2 RHR Service Water Booster pumps, 2 Service Water pumps, and 1 RHR heat exchanger - with containment spray.

- Case C - Operation of one RHR cooling loop with 1 RHR pump, 2 RHR Service Water Booster pumps, 2 Service Water pumps, and 1 RHR heat exchanger - with containment spray.

- Case D - Operation of one RHR cooling loop with 1 RHR pump, 2 RHR Service Water Booster pumps, 2 Service Water pumps, and 1 RHR heat exchanger - no containment spray.

- Case E - Operation of one RHR cooling loop with 1 RHR pump, 1 RHR Service Water Booster pump, 1 Service Water pump, and 1 RHR heat exchanger - with containment spray.

- Case F - Operation of one RHR cooling loop with 1 RHR pump, 1 RHR Service Water Booster pump, 1 Service Water pump, and 1 RHR heat exchanger using suppression pool cooling.

RAI-3

Provide the calculated peak primary containment temperatures and pressures for the design-basis accident (DBA) LOCA and SSLB for CNS. For each case, discuss whether credit was taken for the RHR containment spray system.

NPPD Response:

The maximum accident pressures/temperatures for the drywell are listed below:

- Design Basis Accident (original analysis): 46.2 pounds per square inch-gauge (psig) / 295 degrees Fahrenheit (°F)
- NEDO-10320 Accident Analysis: 58.2 psig
- Accepted value of peak calculated containment pressure for 10 CFR Part 50, Appendix J leak rate testing: 58 psig
- Maximum Extended Load Line Limit/Increased Core Flow (MELLL/ICF): 54.4 psig / 301.4°F

Note that the peak containment pressure of 58.2 psig was determined using the methodology of NEDO-10320. This methodology has been confirmed to be excessively conservative. The current licensing basis calculation yields a more realistic, yet conservative, value of 54.4 psig. Based on this, 58 psig is considered to be a very conservative value of peak containment pressure following a postulated LOCA.

Note also that the peak drywell temperature of 301.4°F is maintained for a short period of time and does not raise the structural drywell temperature above the design value of 281°F.

The above listed pressures and temperatures were calculated early in the event, prior to RHR containment cooling being initiated. Long-term analyses were performed to evaluate containment response for the various cases listed in Item 2, above. The maximum drywell pressure/temperature results for the limiting Cases E and F are listed below:

- Case E (RHR containment spray): 14.9 psig / <300°F
- Case F (RHR suppression pool cooling): 22.7 psig / <300°F

Note that the peak drywell pressures presented for Cases E and F represent secondary peaks that occur after the maximum pressure calculated in the short-term analysis.

The following is a summary table of key results from the SSLB analysis. Results are presented for cases with and without crediting High Pressure Coolant Injection (HPCI) operation.

- Case A-1: 0.01 ft^2 steam line break, with containment sprays at 1200 seconds
- Case A-2: 0.05 ft^2 steam line break, with containment sprays at 600 seconds
- Case A-3: 0.10 ft^2 steam line break, with containment sprays at 600 seconds
- Case A-4: 0.25 ft^2 steam line break, with containment sprays at 600 seconds
- Case A-5: 0.50 ft^2 steam line break, with containment sprays at 600 seconds
- Case A-6: 1.00 ft^2 steam line break, with containment sprays at 600 seconds

Table 1 - SSLB with HPCI

Case	Peak Drywell Pressure (psig)	Peak Drywell Temp. (°F)	Peak Drywell Wall Temp. (°F)
0.01 SSLB Case A-1	16.88	291.7	237.8
0.05 SSLB Case A-2	23.62	320.5	261.8
0.1 SSLB Case A-3	24.82	326.1	266.1
0.25 SSLB Case A-4	26.62	326.0	269.2
0.50 SSLB Case A-5	27.43	329.8	270.3
1.0 SSLB Case A-6	28.14	331.8	271.1

Table 2 - SSLB without HPCI

Case	Peak Drywell Pressure (psig)	Peak Drywell Temp. (°F)	Peak Drywell Wall Temp. (°F)
0.01 SSLB Case A-1	18.86	288.7	246.0
0.05 SSLB Case A-2	24.89	312.1	265.8
0.1 SSLB Case A-3	25.51	328.0	267.2
0.25 SSLB Case A-4	26.05	332.9	268.1
0.50 SSLB Case A-5	26.50	331.7	268.7
1.0 SSLB Case A-6	26.97	332.1	269.8

RAI-4

Compare calculated peak primary containment temperatures and pressures for the DBA LOCA and SSLB with the design limits for CNS, and demonstrate that the calculated values are within the design limits. Justify that the margin between the calculated values and the design limit is acceptable, considering the uncertainties in the plant-specific model used in this case.

NPPD Response:

The maximum allowable drywell pressure and the peak drywell pressures for a LOCA and SSLB are listed below:

- Maximum Code Allowable Internal Pressure: 62 psig
- NEDO-10320 Accident Analysis: 58.2 psig
- MELLL/ICF: 54.4 psig
- SSLB (Case A-6 with HPCI): 28.14 psig

The peak containment pressure of 58.2 psig (accepted as 58 psig by the U.S. Atomic Energy Commission) was determined using the methodology of NEDO-10320. This methodology has been confirmed to be excessively conservative. The current licensing basis calculation yields a more realistic, yet conservative, value of 54.4 psig. Based on this, 58 psig is considered to be a very conservative value of peak containment pressure following a postulated LOCA. Despite the added calculational conservatism, there is still 6 percent margin between the calculated peak pressure of 58.2 psig and the maximum allowable pressure of 62 psig. Additionally, the SSLB analysis results demonstrate that the DBA-LOCA is the limiting event for determining the peak drywell pressure.

The drywell design temperature and maximum drywell temperatures for a LOCA and SSLB are presented below:

- Design Temperature: 281°F
- MELLL/ICF: 301.4°F
- SSLB Peak Drywell Temperature (Case A-6 without HPCI): 332.1°F
- SSLB Peak Drywell Wall Temperature (Case A-6 with HPCI): 271.1°F

The MELLL and ICF analysis, with improved vessel modeling for calculating the blowdown flow rates and flow enthalpies, gives a peak drywell temperature of 301.4°F. This peak value was obtained for the power/flow point of 102%P/75%F (MELLL point). The peak drywell temperature of 301.4°F is maintained for a short time and does not raise the structural drywell temperature above the design value of 281°F.

The results presented above demonstrate that the SSLB is the limiting event for drywell temperature. As detailed in RAI-1, the SSLB analysis was performed with GEs SHEX-04V computer code. The use of the SHEX-04V code has been accepted by the NRC for calculating the response of the containment during an accident or a transient event, and has been applied to

the evaluation of containment response for many BWR plants. NRC letter from A. Thadani to G.L. Sozzi (GE), July 13, 1993, "Use of SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," summarizes the licensee expectation when utilizing the SHEX computer code. Basically, a bench mark of the computer code to an NRC reviewed analysis is required. This bench mark analysis was completed and documented in General Electric report GE-NE-T23-0786-00-01, "Cooper Nuclear Station Containment Analysis Project - NPSH." The bench mark analysis utilized the CNS FSAR Case C for the purpose of SHEX-04V validation and showed that SHEX-04V yielded similar or conservative results compared to the original Case C FSAR analysis.

The inputs for the SSLB analysis were taken from GENE DRF T23-00786-00, "Cooper Nuclear Station - Containment Evaluation - Resolved OPL-4A," dated February 13, 2001. The input values specified in the OPL-4A were selected to produce conservative results in the CNS plant-specific containment analyses. The SSLB analysis contains several sensitivity cases utilized to assess the impact of certain input parameters; such as, containment structural response without containment spray, operator manual initiation time, spray nozzle height, and containment volume.

Containment Structural Temperature without Containment Spray:

This portion of the analysis consisted of the utilization of suppression pool cooling mode of RHR instead of containment spray for mitigation of the SSLB. The 1.0 square foot break size was utilized in this portion of the analysis, since this break size resulted in the highest peak drywell liner temperatures for the spectrum of break sizes analyzed. This showed that the peak drywell shell temperature exceeds the design value, without containment spray. The analysis predicted drywell shell (liner) temperature was 287.1°F. Results from this analysis document the need for containment spray to mitigate temperature effects of a SSLB to maintain the containment structural temperature below the design value of 281°F and quantify the magnitude of the increase above the design temperature, which may be expected without containment spray.

Operator Manual Initiation Time:

The CNS licensing basis maintained in the USAR for manual operator action time for containment cooling functions is 10 minutes. This is maintained in the SSLB analysis; however, a sensitivity analysis was performed to document the results of delaying operator response time to 30 minutes. It is not the intent of this sensitivity analysis to establish new operator response times, nor does it imply that operators would not be able to achieve the licensing basis response time of 10 minutes. Rather this is considered a proactive analysis approach to address any potential future industry issues associated with operator response time for actuation of containment spray. The 0.5 and 1.0 square foot breaks were analyzed with 30 minute response time for manual operator action to initiate containment spray. The results of this sensitivity analysis demonstrate that the peak drywell shell temperature increases to 275.3°F compared to approximately 271°F. This demonstrates that even with 30 minute operator response time, the drywell shell design temperature is not exceeded.

Spray Nozzle Height:

The spray efficiency is a function of air mass to vapor mass ratio, as used by GE in the SHEX code. Therefore, the average vertical drop distance is not used in the analysis to determine spray efficiency. The SHEX model uses this input thermal efficiency to determine the heat transfer from the spray to the drywell atmosphere during the first phase of spray heat transfer to the drywell. Most of the spray heat transfer occurs during this initial phase. The vertical drop distance is used to determine the drop transit time. The drop transit time is used to determine the heat transfer during the second phase of the spray heat transfer to the drywell. During the second phase, heat transfer occurs via convection from the water droplet to the drywell atmosphere. The heat transfer during this time is less significant. Therefore, the effect of vertical drop distance was not expected to impact the analysis results. Regardless, a sensitivity study was performed to compare the assumed 39 foot average drop distance to an assumed 5 foot average drop distance. This was done to demonstrate that equipment interference with the spray distribution and drop distance have negligible impact on the analytical results. A re-analysis of the 0.25 square foot break shows no impact on the results due to a vertical drop distance of 5 feet vice 39 feet.

Containment Volume:

The drywell volume utilized in the analysis, 132,250 cubic feet, is consistent with USAR Table V-2-1. However, due to the near impossibility of obtaining an exact volume for the drywell, a sensitivity analysis was performed to quantify the impact. The sensitivity does not invalidate the licensing basis containment volume reflected in the USAR and utilized in the accident analysis. It was performed to bound the uncertainties of containment volume with respect to the analytical results and demonstrate that calculated drywell response is adequate. A drywell volume of 141,737 cubic feet was utilized for the sensitivity analysis based upon NEDC 98-042, "Estimate of Containment Volumes." A volume of 141,737 cubic feet represents the maximum volume of containment accounting only for major equipment, and is an estimate to compare the adequacy of the historical value. NEDC 98-042 calculates a drywell free volume of 141,737 +0% -10% cubic feet. The 1.0 square foot break was re-analyzed at the higher drywell volume. The results indicate that the peak drywell pressure increased from 28.1 psig to 29.8 psig and the peak drywell airspace temperature increased from 331.8°F to 332.8°F. All increases did not result in approach of design limits and, for all practical purposes, are insignificant. The pressure increase can be attributed to the larger amount of non-condensible gases transferred to the wetwell. Since wetwell volume is not changed, this results in a higher pressure in the drywell prior to vacuum breaker opening for mass redistribution between the wetwell and drywell. This demonstrates that the historical licensing basis value for drywell free air space volume yields acceptable results.

The containment volume sensitivity case demonstrates that it is conservative to assume initially lower drywell temperatures for structural and air space temperature response. The SSLB analysis assumes an initial drywell temperature of 135°F, consistent with the DBA-LOCA analyses contained in USAR Chapter XIV. The initial airspace temperature establishes the non-condensible mass contained in the drywell. Assuming an initial drywell temperature of 135°F vice 150°F results in an initial non-condensible mass approximately 2.5% greater than if 150°F

was used as an initial condition. The sensitivity analysis for containment volume results in an approximate increase of 7.2% in initial non-condensible mass due to the larger drywell volume considered.

The larger drywell initial non-condensible mass obtained by assuming an initial temperature of 135°F versus 150°F results in a higher non-condensible gas-to-steam ratio. This reduces the condensation heat transfer coefficient and, in turn, reduces the rate of heat transfer from the drywell air space to the drywell liner. This, in combination with the resulting higher pressures, leads to higher drywell airspace temperatures, as demonstrated in the sensitivity analysis. The higher drywell pressure can be attributed to the transfer of more non-condensible gas mass to the same wetwell volume, thus creating a larger back pressure for mass transfer through the downcomers. Therefore, use of an initial drywell temperature of 135°F is conservative.

This sensitivity analysis also demonstrates that assuming an initial drywell temperature of 135°F versus 150°F is conservative for determining drywell liner temperature response. Although the condensation heat transfer coefficient is slightly reduced, the larger mass of non-condensibles results in higher drywell pressures. The pressure response is shown to be the controlling factor in the sensitivity analysis. This is due to the increase in the vapor saturation temperature. Therefore, assuming a lower initial drywell temperature results in a slight increase in the predicted peak liner temperature and is conservative.

In summary, the SSLB analysis demonstrates that sufficient margin exists to the maximum allowable drywell pressure and the design drywell temperature. The analysis results were conservatively obtained using a computer code and methodology accepted by the NRC. The analysis input parameters were chosen to conservatively maximize the containment pressure and temperature response. Additionally, sensitivity studies were performed that account for analysis uncertainties.

Attachment 2

Supplement to License Amendment Request to Revise Technical Specifications to Add Residual Heat Removal (RHR) Containment Spray Flowrates

Description

The license amendment request submitted on January 15, 2015 (ML15021A127) contains a discussion of the design minimum spray flowrates of the RHR Containment Spray System and the minimum spray flowrates required to mitigate the small steam line break event (page 7 of 14, first paragraph). The NEDC 00-49 discussed in the referenced paragraph has been recently revised to account for the effects of Diesel Generator low frequency and low voltage conditions. The result of the calculation revision is the design minimum flowrates are slightly lower. These slightly lower flowrates still provide adequate margin to the flow rates required to mitigate the effects of a postulated small steam line break.

Underlining used to highlight values affected by the calculation revision.

What was provided in the January 15, 2015 license amendment request:

> "With concurrent drywell and suppression pool spray operation, the calculation shows the minimum total containment spray flow rate will be <u>7177</u> gallons per minute (gpm), with <u>6480</u> gpm drywell spray and <u>697</u> gpm for suppression pool spray."

Results of Calculation Revision:

> "With concurrent drywell and suppression pool spray operation, the calculation shows the minimum total containment spray flow rate will be <u>6987</u> gallons per minute (gpm), with <u>6302</u> gpm drywell spray and <u>685</u> gpm for suppression pool spray."

The revisions to the design minimum flowrates neither impacts the conclusions of the no significant hazards consideration evaluation that was performed pursuant to 10 CFR 50.91(a)(1), nor the environmental consideration performed pursuant to 10 CFR 51.22.