November 9, 2009

LICENSEE: Nebraska Public Power District

FACILITY: Cooper Nuclear Station Power Plant

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON AUGUST 6, 2009, AND SEPTEMBER 2, 2009, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION STAFF AND NEBRASKA PUBLIC POWER DISTRICT, RELATED TO A CLARIFICATION FOR RESPONSES TO CERTAIN REQUESTS FOR ADDITIONAL INFORMATION, FOR COOPER NUCLEAR STATION LICENSE RENEWAL

The U.S. Nuclear Regulatory Commission staff and representatives of Nebraska Public Power District (the applicant) held telephone conference calls on August 6, 2009, and September 2, 2009, to discuss clarifications of responses to certain requests for additional information for the Cooper Nuclear Station license renewal.

Enclosure 1 provides a listing of the participants, and Enclosure 2 contains a brief description of the conference calls.

The applicant had an opportunity to comment on this summary.

\RA

Bennett Brady, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
As stated

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NAME	SFigueroa	BBrady	TTran	BPham	BBrady (Signature)
DATE	10/28/09	11/09/09	11/09/09	11/09/09	11/09/09

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Letter to Nebraska Public Power District from Tam Tran dated November 9, 2009

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON AUGUST 6, 2009, AND SEPTEMBER 2, 2009, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION STAFF AND NEBRASKA PUBLIC POWER DISTRICT, RELATED TO A CLARIFICATION FOR RESPONSES TO CERTAIN REQUESTS FOR ADDITIONAL INFORMATION, FOR COOPER NUCLEAR STATION LICENSE RENEWAL

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W. Walker (RIV)
G. Pick (RIV)
G. Miller (RIV)

LIST OF PARTICIPANTS
TELEPHONE CONFERENCE CALLS
COOPER NUCLEAR STATION
LICENSE RENEWAL APPLICATION

August 6, 2009

PARTICIPANTS	AFFILIATIONS
Bennett Brady	U.S. Nuclear Regulatory Commission (NRC)
Seung Min	NRC
Jim Davis	NRC
Dave Bremer	Nebraska Public Power District (NPPD)
Jim Loynes	NPPD
Ken Thomas	NPPD
Dave Lach	Entergy (Entergy Nuclear Operations, Inc. (Entergy)
Alan Cox	Entergy
Jacque Lingefelter	Entergy

September 2, 2009

PARTICIPANTS	AFFILIATIONS
Tam Tran	NRC
Seung Min	NRC
Farhad Farzam	NRC
Bill Victor	NPPD
Jim Loynes	NPPD
Ken Thomas	NPPD
Dave Lach	Entergy
Alan Cox	Entergy
Jacque Lingenfelter	Entergy
Reza Ahrabli	Entergy

COOPER NUCLEAR STATION
LICENSE RENEWAL APPLICATION
(Brief description of the conference calls)

RAI 3.1.2.1-2

Clarification requested:

In the applicant's letter to the U.S. Nuclear Regulatory Commission (NRC) dated on October 17, 2000, regarding Title 10 of the *Code of Federal Regulations*, Part 50.59(b)(2) (10 CFR 50.59(b)(2)), Summary Report (page 21), the applicant stated that sampling lines and fittings necessary to hook up analysis hardware were added downstream of existing sample valves on the control rod drive (CRD) suction lines to assist station chemistry personnel in determining the cause for the premature clogging of the CRD suction filters and, following analysis, the sample flow is directed to the reactor building equipment drain system.

Clarify whether the operating experience with the premature clogging of the CRD suction filters has indicated any adverse effect on the piping and piping welds of the CRD system in terms of managing stress corrosion cracking. In addition, confirm, using the operating experience and inspection results, whether stress-corrosion cracking (SCC) has been observed from the piping and components of CRD system or not.

The applicant provided the following clarifications:

Nebraska Public Power District (NPPD) stated that review of operating experience (OE) related to CRD system revealed no occurrences of SCC in CRD system components. Clogging of CRD filters events were not indicative of adverse affects from SCC.

Clarification requested:

In license renewal application (LRA) Table 3.3.2-2 (page 3.3-73) for the CRD system, no plant-specific note for the One-Time Inspection Program is described for the stainless steel piping aging management review (AMR) item, which is exposed to treated water > 140 °F and is susceptible to SCC, although Table 1 Item 3.3.1-38 associated with the stainless steel piping AMR item indicates that the One-time Inspection Program is credited to verify the Water Chemistry Control – Boiling-Water Reactor (BWR) Program for the aging management of the AMR item.

Clarify whether a one-time inspection will be performed on the piping item of the CRD system to confirm that no stress corrosion crackling is occurring in the system or the aging effect of SCC is occurring very slowly.

The applicant provided the following clarifications:

NPPD stated that although not specifically annotated in the Cooper Nuclear Station (CNS) LRA Table 3.3.2-2, the CRD system will undergo one time inspection to verify that the aging effects are not present (same as with other BWR-Water Chemistry Control). It was not cited in the Generic Aging Lessons Learned (GALL) Report; therefore, in order to mirror GALL it was not cited in LRA Table 3.3.2-2.

Clarification requested:

Clarify whether or not the applicant's OE, inspection results or engineering evaluation indicates that the observed premature clogging is related with water chemistry issues that might adversely affect other systems interacting with the CRD system (for example, if applicable, reactor water cleanup [RWCU] system and so on) in terms of managing SCC.

The applicant provided the following clarifications:

NPPD stated that the OE for the CRD did not reveal occurrences where filter clogging was related to water chemistry issues. NPPD stated initially that CRD does not interface with RWCU but later corrected the position and stated that CRD does supply 1-2 gpm for RWCU pump seal cooling and flushing water. However, no indications of SCC have ever been indicated either in the CRD system or the RWCU system that it interfaces.

RAI 3.3.1-1

Clarification requested:

Regarding the RWCU system

RAI response supplement of the applicant by email letter dated on July 30, 2009:

NPPD re-reviewed the examination data sheets for the RWCU welds and could find no reference to a fabrication flaw. There were 13 welds examined in 1989 that had recordable indications evaluated as geometry. They are RWCU-10, RWCU-29, RWCU-30, RWCU-35, RWCU-36, RWCU-37, RWCU-40, RWCU-41, RWCU-42, RWCU-44, RWCU-45, RWCU-47, and RWCU-48. There were two welds that were repaired due to intergranular stress-corrosion cracking (IGSCC) indications, RWCU-13 and RWCU-26. They were last examined in 1995 with no reportable indications. (Note: The applicant's response supplement regarding a flaw described in letter from G. A. Trevors to the NRC, "Generic Letter (GL) 88-01, Cooper Nuclear Station," dated October 9, 1990).

Confirm that no record or reference is available regarding the evaluation of the potential fabrication flaw described in the letter dated October 9, 1990.

The applicant provided the following clarifications:

NPPD responded that they have found no mention of a fabrication flaw in the examination records or in the work orders for the repairs.

Clarification requested:

Clarify whether the OE with the RWCU system indicates that the flaw described in the letter dated October 9, 1990, has not caused leakage or any other adverse effect on the functions of the reactor water cleanup system. In addition, clarify whether additional IGSCC indications have been observed or not since the observation of the two IGSCC indications of the RWCU-13 and RWCU-26 welds.

The applicant provided the following clarifications:

NPPD responded that no further evidence of IGSCC in the non-code RWCU piping has been observed. The last time these welds were ultrasonically examined was in 1995.

Clarification requested:

The response supplement is applicable/relevant for the outboard piping; especially outboard Category D (non-resistant material) portion.

The applicant provided the following clarifications:

NPPD stated, “Yes, the response is only applicable to the non-safety related RWCU piping.”

Clarification requested:

As for the RWCU-13 and RWCU-26 welds from which IGSCC indications were found, clarify how the "repair" was conducted.

The applicant provided the following clarifications:

NPPD stated that a review of MWRs 89-2182 and 89-2183 indicates that the indications were ground out, PT examined, weld repaired and then RT and ultrasonic (UT) examined. The repair welds were not weld overlays.

Clarification requested:

Clarify whether the 1995 inspections are the last inspections on the outboard piping.

The applicant provided the following clarifications:

NPPD stated, "Yes, 1995 was the last year when these examinations were performed. The examination method was UT. After our letter of January 2, 1997, examination of the nonsafety-related RWCU piping was no longer required."

Clarification requested:

What is the piping material associated with the outboard non-resistant welds? What is the weld material of the non-resistant welds in the outboard portion?

The applicant provided the following clarifications:

NPPD stated that the base material is 304 stainless steel. The weld material is 308.

Clarification requested:

What weld materials were used for the repairs of RWCU-13 and RWCU-26 welds? (For example, Inconel ____ or 308L SS, delta ferrite content ____)

The applicant provided the following clarifications:

NPPD stated, "ER308L, delta ferrite content was 9.3 FN (MagnaGage)."

Clarification requested:

1. When were the repairs performed?

   The applicant provided the following clarifications:

   NPPD stated September 1989.

2. Was a stress improvement process such as induction heating stress improvement performed?

   The applicant provided the following clarifications:

   NPPD stated that no additional mitigation measures were applied to these weld repairs.

3. If it exists, provide the references for the applicant's communication with the NRC on the weld repair.

   The applicant provided the following clarifications:

   NPPD stated that no specific communication on these repairs was located.

4. When was the hydrogen water chemistry (HWC) applied to the reactor coolant system?

   The applicant provided the following clarifications:

   NPPD stated that HWC was effectively implemented in August 2003. Noble Metal Chemical Application (NMCA) was first implemented in March 2000. A second application was performed in January 2005.

5. Describe how effective the HWC is to mange SCC in the outboard piping of the reactor water cleanup system.

   The applicant provided the following clarifications:

   NPPD stated that Electrochemical Corrosion Potential (ECP), the measure of HWC effectiveness, is measured in the RWCU system. The ECP probes, located in the Mitigation Monitoring System skid, are supplied from the discharge of the RWCU pumps. The BWRVIA Radiolysis model shows the molar ratio of the water supplied to RWCU at >3:1. This is in agreement with the measured ECP of -490 mV SCE. Therefore, HWC is just as effective for RWCU as it is for Rx Recirculation system.

Clarification requested:

The following references indicate that the applicant found flaws in eight welds: one IGSCC Category A weld (Weld number CSB-BF-1 in the core spray system) and seven IGSCC Category D welds (Weld Nos. RWCU-13, RWCU-26, CWA-CF-37, CWA-CF-38, CWA-CF-40, CWA-CF-43, and CWA-CF-44). The applicant already stated that welds RWCU-13 and RWCU-26 are located in the outboard portion of the reactor water cleanup system and the IGSCC indications in the welds were repaired.

- Letter from G. A. Trevors, NPPD to the NRC, “Request for Additional Information – Generic Letter 88-01 (TAC No. 69131),” July 24, 1989
- Letter from the NRC to G. A. Trevors, NPPD, “Review of Nebraska Power District’s Response to Generic Letter 88-01 (TAC. No. 69131),” February 14, 1990, including Enclosure 2 (pages 11-14)

1. Clarify where the other flaws in Welds CSB-BF-1, CWA-CF-37, CWA-CF-38, CWA-CF-40, CWA-CF-43, and CWA-CF-44 were located (ex. Inboard or outboard) and clarify which Code Class (such as Class 1, 2 or 3) is applied to the piping welds.

The applicant provided the following clarifications:

NPPD provided the following information:

CSB-BF-1	Inboard	Class 1
CSB-CF-37	Outboard	Non-Code
CSB-CF-38	Outboard	Non-Code
CSB-CF-40	Outboard	Non-Code
CSB-CF-43	Outboard	Non-Code
CSB-CF-44	Outboard	Non-Code

2. Clarify whether the flaw-found welds (Welds CSB-BF-1, CWA-CF-37, CWA-CF-38, CWA-CF-40, CWA-CF-43, and CWA-CF-44) were replaced with a SCC-resistant material.

   The applicant provided the following clarifications:

   NPPD stated that the above welds have been replaced with IGSCC resistant materials.

3. If the flaw-found welds (Welds CSB-BF-1, CWA-CF-37, CWA-CF-38, CWA-CF-40, CWA-CF-43, and CWA-CF-44) were repaired, provide the information regarding the repairs including, as applicable, the repair method, piping material, repair weld material, stress improvement processes and when the repairs were conducted.

   The applicant provided the following clarifications:

   NPPD stated that CSB-BF-1 nozzle to safe-end weld and nozzle safe-end were replaced with IGSCC resistant material: The nozzle safe-end base material is 316L. The CSB-BF-1 dissimilar metal weld is Inconel 82 with an Inconel 82 Corrosion Resistant Cladding over the Inconel 182 butter on the reactor vessel nozzle. Weld preparation geometry followed Electric Power Research Institute recommendations to minimize weld induced stress, and the counterbore was polished to remove machine cold working. The welding was closely monitored to control interpass temperatures, heat input and deposition rate. In addition, induction heating stress improvement was performed.

   Furthermore NPPD stated that CWA-CF-37, CWA-CF-38, CWA-CF-40, CWA-CF-43, and CWA-CF-44 were eliminated by the piping replacement in 1990. IGSCC-resistant material, 316L, was used for the replacement. No additional mitigation measures were applied.

4. Clarify what inspections are being performed on the welds in the applicant's aging management programs (AMPs): In addition, clarify what IGSCC categories are currently assigned to the welds in accordance with GL 88-01, as applicable, and whether GL 88-01 inspections are conducted on these welds.

The applicant provided the following clarifications:

NPPD stated that CSB-BF-1 is IGSCC Category A and is included in the scope of the RI-ISI Program.

Furthermore NPPD stated that the CWA welds were previously considered as IGSCC Category D welds. GL 88-01 examinations are no longer required for these welds.

Clarification Requested

Please clarify whether the weld CSB-BF-1 is a BF weld under the ongoing GL-88- 01 inspections.

The applicant provided the following clarifications:

CSB-BF-1 is not examined in the 4th 10-year in-service inspection (ISI) Interval because it is not selected for volumetric examination in accordance with our Risk-Informed ISI Program that subsumed the GL 88-01 Program. The weld is a GL 88-01 Category A weld and, in the current inspection interval, only receives a VT-2 examination each outage under the Pressure test. However, the weld is included in the one-hundred-six GL 88-01 Category A welds under the augmented inspections of B-F and B-J welds as addressed below in relation to RAI B.1.7-5. The CSB-BF-1 weld is a Class 1 Examination Category B-F weld and the current component designation of the weld is CSB-BF-1x. In terms of configuration, the weld is a nozzle-to-safe-end weld of the core spray system loop B.

Clarification requested:

NPPD has indicated that IGSCC-induced cracking has occurred previously in RWCU welds RWCU-13 and RWCU-26, which are located in the non-reactor coolant pressure boundary portions of the RWCU system. NPPD has indicated that although some of the non-reactor coolant pressure boundary portions of the RWCU system have been replaced with IGSSC-resistant material, not all of the piping has been modified to use the resistant material. The past occurrence of IGSCC-induced indications in these welds demonstrates that IGSCC is a credible age-related cracking mechanism for the portions of the system that are designed with the original non-resistant material. NPPD continues to credit only the Water Chemistry Control – BWR Program to manage cracking in this piping as a result of IGSCC, when coupled to the implementation of a one-time inspection for the outboard piping components in the system.

GALL AMP XI.M32, “One-Time Inspection,” states that the One-Time Inspection Program is a valid program to credit for cases where confirmation is needed to confirm that either an aging effect is indeed not occurring or that an aging effect is occurring very slowly so as not to affect the component or structure intended function during the period of extended operation.

In consideration of this OE, justify why the crediting of the Water Chemistry Control – BWR Program with a one-time inspection is considered to be capable of managing IGSCC-induced cracking in the portion of the system that has not been replaced with the IGSCC-resistant material, given the fact the IGSCC has occurred in the system in the past.

Justify why periodic inspections are not credited for the active degradation mechanism in the outboard portion of the RWCU system.

The applicant provided the following clarifications:

NPPD stated that per GL 88-01 Supplement 1, examination of the RWCU piping outside of containment is no longer required.

NPPD stated that per LRA Table 3.3.3, Item 3.3.1-37, CNS has replaced portions of the RWCU system piping with material that is not susceptible to IGSCC. CNS has complied with the requirements of NRC GL 89-10 and has performed the inspections specified by NRC GL 88-01 with no significant indications of IGSCC on piping that was not replaced. The Water Chemistry Control – BWR Program with a one-time inspection Water Chemistry Control – BWR Program is believed to be adequate for managing the aging affects.

RAI B.1.7-3 and RAI B.1.7-6:

Clarification requested:

Regarding non-Class 1 components in terms of crediting the BWR SCC AMP

Except for the RWCU system, confirm whether the engineered safety features and auxiliary systems including the residual heat removal system have non-Class 1 stainless steel piping and piping welds, which have the standby or operating temperature greater than 200 °F such that the components are under the scope of GL 88-01.

The applicant provided the following clarifications:

NPPD stated that there are no stainless steel welds 4 nominal pipe size (NPS) and larger in the safety-related piping systems outside of containment. Thus GL 88-01 does not apply to any class 2 or 3 welds.

Clarification requested:

Confirm whether the applicant's foregoing evaluation is consistent with the on-site aging management review documents for non-Class 1 components.

The applicant provided the following clarifications:

NPPD stated that it is consistent.

RAI 3.1.2.1-1

Regarding the One-Time Inspection Program in terms of managing SCC BWR SCC AMP Scope

Clarification requested:

In relation with LRA Table 1 Items 3.2.1-18 and 3.3.1-38, the applicant stated that the One-Time Inspection Program is used to verify the effectiveness of the Water Chemistry Control – BWR Program.

However, the staff noted that no plant-specific note for the One-Time Inspection Program is associated with any AMR item that is described in LRA Sections 3.2 (engineered safety features) and 3.3 (auxiliary systems) in relation with LRA Table 1 Item 3.2.1-18 and 3.3.1-38, respectively, for managing SCC in conjunction with the Water Chemistry Control – BWR Program.

Clarify how the One-Time Inspection Program implements a one-time inspection on the components of engineered safety features and auxiliary systems to verify the effectiveness of the water chemistry program in terms of managing SCC in the systems.

LRA Table 3.1.2-3 for the reactor coolant pressure boundary (page 3.1-61) addresses stainless steel tubing (non-Class 1) item which involves an environment of "Treated Water > 140 °F," an aging effect of "cracking," an AMP of "Water Chemistry Control –BWR," GALL Vol. 2 Item V.D2-29, LRA Table 1 Item 3.2.1-18, a consistency note of E and a plant-specific note of 105 indicating that the item is less than 4 inches NPS and is not the part of the reactor coolant pressure boundary.

The staff also noted that similarly a non-Class 1 item of piping and fittings is addressed in Table 3.1.2-3 (page 3.1-57) with the same combination of LRA Table 2 attributes as the non-class 1 tubing item.

Clarify why the non-Class 1 stainless steel tubing and piping items in Table 3.1.2-3, associated with stress corrosion cracking, are listed in the reactor coolant pressure boundary table which is regarded to mainly address Class 1 items.

The applicant provided the following clarifications:

NPPD stated that per LRA App-B.1.29, the One-Time Inspection Program is accredited for all line items that cite a water chemistry control program. However, a plant-specific note was not applied consistent with no note cited in the GALL.

Aging management reviews are done on a system basis. For practicality, Table 3.1.2-3 presents the aging management review results of both the Class 1 reactor coolant pressure boundary components (most of which are from the reactor coolant system) and the non-Class 1 components of the reactor coolant system.

RAI B.1.7-5

Clarification requested:

Regarding RI-ISI and GL 88-01 Inspection requirements

Provide the following information on the RI-ISI inspections for Category A welds in order to confirm whether the applicant's inspections are consistent with the requirements of GL 88-01.

The applicant provided the following clarifications:

Water chemistry:

NPPD stated that, although CNS meets the BWRVIP Water Chemistry guidelines for HWC and NMCA, there has been no change to the examination frequency for Category A welds in the RI-ISI Program.

Inspection extent and schedule (minimum):

NPPD stated that 25% of the Category A welds in the RI-ISI Program are examined during the 10-year interval. This is consistent with both GL 88-01 and BWRVIP-75-A requirements.

Clarification requested:

Regarding RAI B.1.7-5 which was issued to the applicant on June 29, 2009 (ML091600284), and responded by the applicant on July 29, 2009 (ML092160083), the staff asked for further clarification on the implementation of the criteria for discontinuation of the RWCU system outboard piping inspections. The staff requested the following clarifications:

- A comparison was made between the applicant's response to the question regarding the inspection extent (minimum):

  "NPPD stated that 25% of the Category A welds in the RI-ISI Program are examined during the ten-year interval. This is consistent with both GL 88-01 and BWRVIP-75-A requirements and information contained in the applicant's 4-th interval ISI plan (NLS2008093, Enclosure 1 of USAR Revision XXIII). However, Cooper Station 4th Interval In-service Inspection) which indicates the inspection extent of GL 88-01 Category A welds is regarded as 18 welds out of 106 welds [or ~ 17%] rather than 25 % stated above."

Explain why there is a discrepancy between 17% and 25%.

The applicant provided the following clarifications in August 27, 2009 email:

The requirements of GL 88-01 for Inspection Schedules and Sample Expansion were superseded by BWRVIP-75-A, which has been approved by the NRC. The CNS RI-ISI Program meets BWRVIP-75-A for Inspection Schedules, Sample size for Inspection, and Sample Expansion. For GL 88-01 Category A welds, the RI-ISI Program examines 25% of the Category B-F welds over the ten-year interval and 10% of the Category B-J welds. (No credit is assumed for HWC.) Note that:

- There are 21 Category A B-F welds @ 25% = 6, we scheduled 8 for examination
- There are 85 Category A B-J welds @ 10% = 9, we scheduled 10 for examination
- Accordingly, there are a total of 18 welds to be examined out of a population of 106 Category A's.

Clarification requested:

In response to the above response provided in the August 27, 2009 email*:*

Note 3(b) of Table 3-1 in BWRVIP-75-A (approved BWRVIP-75) states that:

> "Category B-J welds (in Category A) can be inspected within a scope of 10% every 10 years when a second mitigator is applied (such as heat sink welding, mechanical stress improvement process and so on)."

Therefore, please, clarify which second mitigator for 10% inspection extent is relevant for Category A B-J welds."

The applicant provided the following clarifications*:*

NPPD indicated the second mitigator is induction heating stress improvement.

Regarding the AMR item for the LPRM

Clarification requested:

Background:

LRA Table 3.1.2-2 addressed one AMR item for incore flux monitor local power range monitors (LPRMs) (page 3.1-49). Cracking is managed by the ISI and water chemistry programs.

Follow-up Request:
Does the LPRM item (LRA page 3.1-49) represent the incore housings of the LPRMs (but the portion below/outside the vessel)? Otherwise, please provide a specific component description for this AMR item.

The applicant provided the following clarifications:

The LPRM item refers to the local power range monitors themselves. The following describes the AMR item in more detail:

> "The local power range monitors (LPRM) have no dry tubes. The detector itself is the pressure boundary. The power range detectors are inserted in the incore guide tubes and are held in place at the top end to the top fuel guide by means of a spring-loaded plunger. Detectors are sealed (bolted) to thimble flanges under the vessel."

The guide tubes are also in Table 3.1.2-2 (reactor vessel internals) while the incore monitor housings are part of the vessel AMR and appear in Table 3.1.2-1.

Cooper Nuclear Station

cc:

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