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54.17

NLS2010044
April 28, 2010

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

Subject: Response to Open and Confirmatory Items from the Safety Evaluation Report Related to the License Renewal of Cooper Nuclear Station
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:
1. Letter from Brian E. Holian, U.S. Nuclear Regulatory Commission, to Stewart B. Minahan, Nebraska Public Power District, dated April 6, 2010, "Safety Evaluation Report With Open Items Related to the License Renewal of Cooper Nuclear Station."
2. Letter from Stewart B. Minahan, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated September 24, 2008, "License Renewal Application" (NLS2008071).

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District to respond to certain open and confirmatory items documented in the Nuclear Regulatory Commission's Safety Evaluation Report related to the Cooper Nuclear Station License Renewal Application (LRA) (Reference 1). This response is provided in Attachment 1. Certain conforming changes to the LRA (Reference 2) are provided in Attachment 2.

Should you have any questions regarding this submittal, please contact David Bremer, License Renewal Project Manager, at (402) 825-5673.

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NRR

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

Executed on 4/28/10
(Date)

Sincerely,

Brian J. O'Grady
Vice President – Nuclear and
Chief Nuclear Officer

/wv

Attachments

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/ attachments

CNS Records w/ attachments

Attachment 1

Response to Open and Confirmatory Items from the
Safety Evaluation Report Related to the License Renewal of Cooper Nuclear Station
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER) related to the License Renewal of Cooper Nuclear Station (CNS) contains four Open Items (OI) and one Confirmatory Item (CI). The Nebraska Public Power District (NPPD) has discussed two of the OIs and the CI with the NRC and believes the dispositions below will provide satisfactory closure. The remaining two OIs will be discussed in future correspondence. The Open and Confirmatory Items are shown in italics, and the NPPD responses are shown in block font.

NRC Open Item: *OI 2.3.4.2-1: (SER Section 2.3.4.2 - Steam and Power Conversion Systems In-Scope for 10 CFR 54.4(a)(2))*

LRA Section 2.3.4.2 describes the steam and power conversion systems within the scope of license renewal in accordance with 10 CFR 54.4(a)(2), which includes the condensate makeup system. During its review of the LRA, staff determined that the condensate storage tank (CST) 1A should have been included within the scope of license renewal in accordance with 10 CFR 54.4(a)(2). The applicant has not agreed to this scoping issue. This is OI 2.3.4.2-1.

NPPD Response:

In response to the open item, NPPD is revising the scoping of the condensate makeup system to conservatively include an intended function corresponding to 10 CFR 54.4(a)(2). CST 1A provides a source of water to core spray pumps and two residual heat removal (RHR) pumps during shutdown operations in the rare situation when the suppression pool is drained in Mode 4 or in Mode 5 with the spent fuel storage pool gates not removed or the water level is less than 21 feet over the top of the reactor pressure vessel flange.

The License Renewal Application (LRA) is amended to include CST 1A and associated piping and valves supplying water to the core spray and RHR pumps. Components are added to the core spray system aging management review.

The following LRA sections are revised to incorporate components performing this intended function (see Attachment 2, Changes 1 through 8):

- Table 2.3.2-2, Core Spray System, Components Subject to Aging Management Review, addition of component type *tank*.
- Section 2.3.4.2, Steam and Power Conversion Systems in Scope for 10 CFR 54.4(a)(2), revise description of Condensate Makeup and add intended function.

- Section 3.2.2.1.2, Core Spray, add three environments and two aging management programs (Aboveground Steel Tanks, Buried Piping and Tanks Inspection).
- Table 3.2.2-2, Core Spray System, Summary of Aging Management Evaluation, add line items for component types *piping* and *tank.*
- Table 3.4.1, Summary of Aging Management Programs for Steam and Power Conversion System Evaluated in Chapter VIII of NUREG-1801, revise two line items.

NRC Open Item: *OI 3.0.3.1-1: (SER Section 3.0.3.1.11 - One-Time Inspection, Small Bore Piping Program)*

LRA Section B.1.30 describes the new One-Time Inspection – Small Bore Piping Program, which the applicant claims to be consistent with GALL AMP XI.M35, "One-Time Inspection of ASME Code Class 1 Small Bore Piping." During its review, the staff determined that small bore piping includes socket welds, and that, because of operating experience with failures of socket welds, a periodic inspection of such welds under a plant-specific program, consistent with GALL AMP XI.M35, would be appropriate at CNS for license renewal. The applicant has not agreed to include socket weld nor commit to a plant-specific program at CNS. This is OI 3.0.3.1-1.

NPPD Response:

The issue of OI 3.0.3.1-1 is whether CNS-specific operating experience with small-bore socket welds warrants a periodic aging management program rather than a one-time inspection. Absent the need for a periodic aging management program, the CNS program is consistent with the program recommended in NUREG-1801 for managing the effects of aging in ASME Code Class 1 small-bore piping including socket welds.

Periodic Versus One-Time Inspection

In response to follow-up questions related to the response to RAI B.1.30-2, NPPD performed additional detailed review of the cause and corrective actions for cracking found in socket welds at CNS. As stated in the response (NLS2009092) to RAI B.1.30-2, the identified cause of the socket weld cracks found at CNS was high-cycle fatigue from excessive vibration unrelated to the effects of aging. The vibration occurred on piping that was in service only during plant shutdown operations. Vibration of piping and components during operation of the 24-inch injection line for RHR system loop "A" resulted in cracks and crack-like indications in associated vent and drain lines. Loose and rotated parts, broken bolts, loose hand wheel, and loose bolts on valve RHR-MOV-27A, Outboard Injection Valve, observed in 1991 are evidence of the high vibration which resulted in the cracked welds. In addition, vibration measurements taken before and after installation of new valve trim in the flow control valve showed a significant reduction in the levels of vibration. A 1993 analysis concluded that the high levels of vibration were induced by throttling RHR-MOV-27A. Previous cracks had been found in the same area of piping; one in 1977 and two in 1991.

As the cause of the vibration was inadequate design, the corrective actions were to modify the design by 1) installation of new valve trim for RHR-MOV-27A and 2) installation of new supports for adjacent drain lines. Additional actions taken included the following:

- Replacement of cracked piping in vent and drain lines.

- Non-destructive evaluation (dye penetrant inspection (PT)) of welds in adjacent vent and drain lines on the 24-inch injection line for RHR system loop "A."
  Results: crack-like indications were identified and repaired in a vent line piping section.

- PT inspections of various welded vent, drain, and test connection piping in loops A and B of the RHR system over the period of October 1994 to April 1997.
  Results: no additional cracking was identified during these PT inspections.

Subsequent to the 1993 experience, during operation of three manual drain valves in December 2003, a small crack was identified in a section of ¾-inch nominal pipe size (NPS) piping. The crack was at the toe of a socket weld in a drain line for RHR system loop "A" 24-inch injection line that had not been replaced in 1993. The cracked weld was removed from the system and sent to a laboratory for analysis.

Laboratory analysis concluded that the crack was initiated by torsional fatigue as a result of the rotation of the three drain valves during the vibration of RHR loop "A." The term torsional fatigue describes the vibration induced loading of the cracked weld fitting caused by the attached manual valves twisting around their center of mass. The amount of corrosion found on the crack surface indicated that this crack had initiated at some time well in the past and gone undetected until 2003.

NPPD performed additional vibration monitoring of the RHR loops following discovery of the crack in 2003. This monitoring identified no excessive vibration during system operation, further supporting the conclusion that the crack had initiated prior to the 1993 corrective actions to ameliorate the vibration on the RHR loop piping.

In summary, the cracking identified in these small-bore piping socket welds at CNS was not related to the effects of aging but was due to design deficiencies that resulted in high levels of vibration in the 24-inch injection line for RHR system loop "A." Design modifications to the system eliminated the cause of the cracking. Piping replaced in 2004 included additional socket welds that had the potential to be affected by the vibration. Destructive examinations of the replaced socket welds showed no cracking as a result of the previous excessive vibration or any other potential aging mechanism. As the cracking was not due to stress corrosion cracking or thermal or mechanical loading, the one-time inspection program remains appropriate for managing the effects of aging on Class 1 small-bore piping.

Aging Management Program

NUREG-1801 Section XI.M35 recommends volumetric examination as the method for one-time inspection of Class 1 small-bore piping. In the absence of a qualified and proven technique for non-destructive volumetric examination of socket welds, NPPD has performed destructive examinations as described below that provide confirmation that cracking due to aging mechanisms is not occurring in Class 1 small-bore socket welds.

To assess the condition of socket welds in the vicinity of the 2003 cracked weld, in 2004 NPPD removed sections of RHR piping containing 12 additional socket welds. The piping containing the 12 socket welds was shipped to an independent laboratory for analysis. The laboratory performed non-destructive and destructive examinations of all 12 socket welds. Two liquid penetrant methods were used to inspect the pipe samples. One inspection method was red-dye liquid penetrant testing and the other method was fluorescent-dye liquid penetrant testing. After penetrant testing, the welds were destructively examined by sectioning the weld areas and performing optical microscopy. No fatigue or stress corrosion cracking was identified in any of the 12 welds.

The One-Time Inspection – Small-Bore Piping Program described in CNS LRA B.1.30 and recommended by NUREG-1801, Section XI.M35, provides reasonable assurance that the effects of aging of Class 1 piping less than four inches NPS will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation. The XI.M35 aging management program recommends one-time volumetric examination of selected weld locations to manage cracking. While not specifically stated in XI.M35, XI.M32 One-Time Inspection recommends that one-time inspections be conducted no earlier than 10 years prior to the period of extended operation, and in such a way as to minimize the impact on plant operations. As a plant will have accumulated at least 30 years of operation before the inspections, sufficient time will have elapsed for aging effects, if any, to be manifest. The laboratory examinations of the 12 socket welds at CNS in 2004 were conducted within the recommended 10-year period prior to the period of extended operation. The 12 socket welds constitute a reasonable sample based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of ASME Code Class 1 small-bore piping socket weld locations as recommended in NUREG-1801, XI.M35, Monitoring and Trending. For socket welds, these examinations fulfill the recommendations of NUREG-1801 XI.M35 for volumetric inspection of a sample of Class 1 small-bore piping welds. For butt-welded piping, the one-time inspection of Class 1 small-bore piping will entail volumetric inspections of butt weld locations prior to the period of extended operation using volumetric inspection techniques with demonstrated capability and a proven industry record to detect cracking in piping weld and base material.

Conclusion

Consistent with GALL AMP XI.M35, "One-Time Inspection of ASME Code Class 1 Small-Bore Piping," the CNS program includes socket welds. A substantial sample of socket welds has been destructively examined. A plant-specific periodic inspection program, above and beyond inspections performed under the Inservice Inspection (ISI) Program, is not warranted at CNS based on plant operating experience. Cracking observed in Class 1 small-bore piping at CNS was directly attributable to inadequate design during the operation of specific RHR throttling valves. The design inadequacies have been corrected eliminating the root cause of the observed cracking. The completed one-time inspections and subsequent operating experience indicate that ongoing Water Chemistry Control - BWR and Inservice Inspection - ISI Programs will be effective in managing the effects of aging on Class 1 small-bore piping socket welds during the period of extended operation.

During the period of extended operation, visual examinations (VT-2) will be conducted on socket weld fittings. The VT-2 examinations will be performed by certified examiners using ASME Section XI approved visual inspection procedures consistent with ASME Section XI.

Notwithstanding the basis justifying that the one-time inspection program is appropriate for CNS Class 1 small-bore socket welds, NPPD will perform periodic volumetric inspection of Class 1 small-bore socket welds during the period of extended operation, as described in the commitment below. In accordance with the NRC-approved risk-informed ISI plan, three socket weld examinations are scheduled for the 2011 refueling outage.

Commitment

During the period of extended operation, NPPD will perform periodic volumetric examinations of Class 1 socket weld connections. Three Class 1 socket welds will receive volumetric examination during each 10 year ISI interval. The examination method will be a volumetric examination of the base metal ½" beyond the toe of the socket fillet weld which allows for the use of qualified ultrasonic examination techniques as close as possible to the fillet weld. The volumetric examinations will be performed by certified examiners following guidelines set forth in ASME Section V, Article 4 consistent with the guidelines for examination volume of ½" beyond the toe of the weld as established in MRP-146, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines."

NRC Confirmatory Item: *CI 4.3.3.2-1: (SER Section 4.3.3 - Effects of Reactor Water Environment on Fatigue Life)*

Section 4.3.3 of the LRA describes the applicant's evaluation of the effects of the reactor coolant environment on the fatigue life of components. In the LRA and subsequent RAI response, the applicant indicated that its fatigue correction (F_{en}) factor for alloy 600 was calculated in accordance with the method described in NUREG/CR 6335. However, the staff noted that NUREG/CR-6909 contains later data and information that provide equations for determining a F_{en} factor that can result in more conservative value than the value calculated by the applicant. The staff requests that the applicant demonstrates adequate conservatism using the NUREG/CR 6335 methodology. This is CI 4.3.3.2-1.

NPPD Response:

NPPD is revising LRA commitment NLS2008071-08 as follows (shown in underline):

"Consideration of the effect of the reactor water environment will be accomplished through implementation of one or more of the following options for the reactor vessel shell and lower head, feedwater nozzles, core spray nozzles and RHR pipe transition.

(1) Update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). [LRA Section B.1.15] NPPD will use NUREG/CR-6909 when determining the effects of the reactor coolant environment on the fatigue life of Alloy 600 components. [CI 4.3.3.2-1]

(2) Repair or replace the affected locations before exceeding an environmentally adjusted CUF of 1.0. [RAI B.1.15-1]

The CNS Fatigue Monitoring Program will be enhanced to require the recording of each transient associated with the actuation of a safety/relief valve (SRV). [LRA Section B.1.15]"

Attachment 2

Changes to the License Renewal Application
Cooper Nuclear Station, Docket No. 50-298, DPR-46

This attachment provides changes to the License Renewal Application (LRA) that conform to the positions taken in Attachment 1. The changes are presented in underline/strikeout format.[1]

1. LRA Table 2.3.2-2, "Core Spray System Components Subject to Aging Management Review," is revised to read:

Component Type	Intended Function(s)
Bolting	Pressure boundary
Cyclone - separator	Pressure boundary Filtration
Flange	Pressure boundary
Flow element	Pressure boundary
Instrument snubber	Pressure boundary
Piping	Pressure boundary
Pump casing	Pressure boundary
Restriction orifice	Pressure boundary Flow control
Strainer	Filtration
<u>Tank</u>	<u>Pressure boundary</u>
Tubing	Pressure boundary
Valve body	Pressure boundary

Reference: Response to OI 2.3.4.2-1.

2. LRA Section 2.3.4.2, "Steam and Power Conversion Systems in Scope for 10 CFR 54.4(a)(2)," Page 2.3-168 under "Condensate Makeup" is revised to read:

"The 450,000-gallon and 700,000-gallon main condensate storage tanks supply the various station requirements. The two tanks can receive demineralized makeup water from the water treatment plant or reprocessed water from the radwaste system with the

[1] The changes shown are made against the original LRA submitted on September 24, 2008. Where other previously made LRA changes affect the same text, a footnote is provided cross-referencing the letter where the previous change was made.

smaller tank providing water to the larger tank. The tanks are constructed of coated carbon steel with electric heaters for anti-freeze protection. The 700,000-gallon tank has a steel retaining wall to prevent spillage from a tank rupture or overflow of radioactive water. The 450,000-gallon main condensate storage tank, CST 1A, can be aligned to supply the core spray pumps and two residual heat removal pumps. CNS Technical Specification Bases B.3.5.2, ECCS—Shutdown, allows the suppression pool to be drained during Mode 4 and Mode 5 provided two operable CS or LPCI subsystems are aligned to take a suction on CST 1A and the CST contains at least 150,000 gallons of water."

Reference: Response to OI 2.3.4.2-1.

3. LRA Section 2.3.4.2, "Steam and Power Conversion Systems in Scope for 10 CFR 54.4(a)(2)," Page 2.3-168 under "Condensate Makeup" is revised to read:

"The CM system has the following intended functions for 10 CFR 54.4(a)(1).

- Provide water to the ECCS systems.

The CM system has the following additional intended function for 10 CFR 54.4(a)(2).

- Provide water to the CS and RHR pumps during shutdown operations with the suppression pool drained."

Reference: Response to OI 2.3.4.2-1.

4. LRA Section 2.3.4.2, "Steam and Power Conversion Systems in Scope for 10 CFR 54.4(a)(2)," Page 2.3-168 under "Condensate Makeup" is revised to read:

"The ECSTs and CM system components that support the HPCI system pressure boundary are reviewed with the high pressure coolant injection system (Section 2.3.2.4). CST 1A and associated piping not identified as in scope for (a)(2) for leakage and spray are reviewed with the core spray system (Section 2.3.2.2). Valves associated with the standby gas treatment system loop seal are reviewed with the standby gas treatment system (Section 2.3.2.6)."

Reference: Response to OI 2.3.4.2-1.

5. LRA Section 3.2.2.1.2, "Core Spray," Page 3.2-4 is revised to read:

"**Environment**

Core spray system components are exposed to the following environments.

- air – indoor
- air – outdoor
- concrete
- soil
- treated water"

Reference: Response to OI 2.3.4.2-1.

6. LRA Section 3.2.2.1.2, "Core Spray," Page 3.2-4 is revised to read:

"**Aging Management Programs**

The following aging management programs manage the effects of aging on the core spray system components.

- Aboveground Steel Tanks
- Bolting Integrity
- Buried Piping and Tanks Inspection
- External Surfaces Monitoring
- Water Chemistry Control – BWR"

Reference: Response to OI 2.3.4.2-1.

7. LRA Table 3.2.2-2, "Core Spray System, Summary of Aging Management Evaluation" is revised to add the following line items:

Piping	Pressure boundary	Carbon steel	Air – outdoor (ext)	Loss of material	External Surfaces Monitoring	V.E-8 (E-45)	3.2.1-31	A
Piping	Pressure boundary	Carbon steel	Soil (ext)	Loss of material	Buried Piping and Tanks Inspection	V.B-9 (E-42)	3.2.1-17	C
Tank	Pressure boundary	Carbon steel	Air – outdoor (ext)	Loss of material	Aboveground Steel Tank	VIII.E-39 (S-31)	3.4.1-20	C
Tank	Pressure boundary	Carbon steel	Concrete (ext)	Loss of material	Aboveground Steel Tank	--	--	G
Tank	Pressure boundary	Carbon steel	Treated water (int)	Loss of material	Water Chemistry Control – BWR	VIII.E-40 (S-13)	3.4.1-6	C, 201

Reference: Response to OI 2.3.4.2-1.

8. LRA Table 3.4.1, "Summary of Aging Management Programs for the Steam and Power Conversions Systems Evaluated in Chapter VIII of NUREG-1801," is revised to read:

3.4.1-6	Steel and stainless steel tanks exposed to treated water	Loss of material due to general (steel only) pitting and crevice corrosion	Water Chemistry and One-Time Inspection	Yes, detection of aging effects is to be evaluated	Consistent with NUREG-1801. Loss of material in steel tanks exposed to treated water is managed by the Water Chemistry Control – BWR Program. The One-Time Inspection Program will be used to verify the effectiveness of the water chemistry program. There are no stainless steel tanks exposed to treated water in the steam and power conversion systems. <u>This line applies to components in the core spray system listed in Table 3.2.2-2 and to components</u> ~~The components to which this NUREG-1801 line item applies are included~~ in scope under criterion 10 CFR 54.4(a)(2) and listed in series 3.4.2-2-xx tables. See Section 3.4.2.2.2 item 1 and Section 3.4.2.2.7 item 1.

3.4.1-20	Steel tanks exposed to air – outdoor (external)	Loss of material/ general, pitting, and crevice corrosion	Aboveground Steel Tanks	No	<u>Consistent with NUREG-1801. The Aboveground Steel Tanks Program will manage loss of material for the steel condensate storage tank exposed to outdoor air. This line applies to components in the core spray system listed in Tables 3.2.2-2.</u> ~~This item was not used. There are no steel tanks exposed to outdoor air with intended functions in the steam and power conversion systems.~~

Reference: Response to OI 2.3.4.2-1.

9. LRA Section A.1.1.15 is revised to read[2]:

"The Fatigue Monitoring Program will be enhanced as follows:

- Consideration of the effect of the reactor water environment will be accomplished through implementation of one or more of the following options for the <u>reactor vessel shell and lower head,</u> feedwater nozzles, core spray nozzles and RHR pipe transition.

[2] The Fatigue Monitoring Program enhancements on Page A-8 were previously changed in NLS2009040 (ADAMS Accession Number ML091690050) in response to RAI B.1.15-1.

(1) Update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). NPPD will use NUREG/CR-6909 when determining the effects of the reactor coolant environment on the fatigue life of Alloy 600 components.

(2) Repair or replace the affected locations before exceeding a CUF of 1.0.

- The CNS Fatigue Monitoring Program will be enhanced to require the recording of each transient associated with the actuation of a safety/relief valve (SRV).

Enhancements will be implemented at least two years prior to entering the period of extended operation."

Reference: CI 4.3.3.2-1

10. LRA Section B.1.15 is revised to read[3]:

Elements Affected	Enhancement
2. Preventive Actions 4. Detection of Aging Effects 6. Acceptance Criteria 7. Corrective Actions	Consideration of the effect of the reactor water environment will be accomplished through implementation of one or more of the following options for the reactor vessel shell and lower head, feedwater nozzles, core spray nozzles and RHR pipe transition. (1) Update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). NPPD will use NUREG/CR-6909 when determining the effects of the reactor coolant environment on the fatigue life of Alloy 600 components. (2) Repair or replace the affected locations before exceeding a CUF of 1.0.

Reference: CI 4.3.3.2-1

[3] The Fatigue Monitoring Program enhancements on Page B-48 were previously changed in NLS2009040 (ADAMS Accession Number ML091690050) in response to RAI B.1.15-1.

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©[4]

Correspondence Number: NLS2010044

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

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
During the period of extended operation, NPPD will perform periodic volumetric examinations of Class 1 socket weld connections. Three Class 1 socket welds will receive volumetric examination during each 10 year ISI interval. The examination method will be a volumetric examination of the base metal ½" beyond the toe of the socket fillet weld which allows for the use of qualified ultrasonic examination techniques as close as possible to the fillet weld. The volumetric examinations will be performed by certified examiners following guidelines set forth in ASME Section V, Article 4 consistent with the guidelines for examination volume of ½" beyond the toe of the weld as established in MRP-146, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines."	NLS2010044-01	January 18, 2014
Consideration of the effect of the reactor water environment will be accomplished through implementation of one or more of the following options for the reactor vessel shell and lower head, feedwater nozzles, core spray nozzles and RHR pipe transition. (1) Update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-	NLS2008071-08 Revision 2	January 18, 2012

approved code case). [LRA Section B.1.15] NPPD will use NUREG/CR-6909 when determining the effects of the reactor coolant environment on the fatigue life of Alloy 600 components. [CI 4.3.3.2-1] (2) Repair or replace the affected locations before exceeding an environmentally adjusted CUF of 1.0. [RAI B.1.15-1] The CNS Fatigue Monitoring Program will be enhanced to require the recording of each transient associated with the actuation of a safety/relief valve (SRV). [LRA Section B.1.15]		