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54.17

NLS2009062
September 24, 2009

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

Subject: Response to Request for Additional Information for the Review of Cooper Nuclear Station License Renewal Application
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:
1. Letter from Tam Tran, U.S. Nuclear Regulatory Commission, to Stewart B. Minahan, Nebraska Public Power District, dated July 29, 2009, "Request for Additional Information for the Review of the Cooper Nuclear Station License Renewal Application (TAC No. MD9763 and MD9737)" (ADAMS Accession Number ML092090276).

2. Letter from Bennett M. Brady, U.S. Nuclear Regulatory Commission, to Stewart B. Minahan, Nebraska Public Power District, August 28, 2009, "Request for Additional Information for the Review of the Cooper Nuclear Station License Renewal Application (TAC No. MD9763 and MD9737)" (ADAMS Accession Number ML092310654).

3. Letter from Stewart B. Minahan, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated September 24, 2008, "License Renewal Application."

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District to respond to the Nuclear Regulatory Commission Requests for Additional Information (RAI) (References 1 and 2) regarding the Cooper Nuclear Station License Renewal Application (LRA). These responses are provided in Attachments 1 and 2, respectively. Certain changes to the LRA (Reference 3) have been made to reflect these RAI responses and other clarifications. These changes are provided in Attachment 3.

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 9/24/09
(Date)

Sincerely,

Stewart B. Minahan
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 3 LIST OF REGULATORY COMMITMENTS©[4]

Correspondence Number: NLS2009062

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
None		

Attachment 1

Response to Request for Additional Information
for License Renewal Application
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) regarding the License Renewal Application is shown in italics. The Nebraska Public Power District's (NPPD) response to this RAI is shown in block font.

NRC Request: *RAI B.1.20-1*

Background:
There have been reoccurring failures of main steam line pipe supports at CNS since the 1980's. In 2006 a Condition Report (CR-CNS-2006-09590, Action Item 8) was initiated to address the recurring pipe support deficiencies with completion of all redesign and associated hardware changes (if necessary) by June 30, 2011.

Issue:
Continued failures of steam line supports during each period of operation between plant outages can affect the structural integrity of the main steam line piping system.

Request:
Explain how the corrective action process will address the potential aging effects (i.e. structural fatigue) on the piping system due to the unanalyzed loading condition associated with the past pipe support failures.

NPPD Response:

Reoccurring failures of Cooper Nuclear Station (CNS) main steam (MS) line pipe supports and the potential effect on the piping system have been identified by NPPD and addressed through the Corrective Action Program (CAP).

The follow up CAP Apparent Cause Evaluation indicated MS pipe support deficiencies are the result of high-cycle fatigue loadings due to normal flow and thermal dynamic operating conditions of the system (not aging-related). A detailed vendor-assisted engineering study of the 24" diameter MS system in the heater bay is necessary to determine modifications to the piping system which may help to eliminate the recurring pipe support deficiencies. The vendor has developed recommendations to resolve the MS line support deficiencies. Plans are in place at CNS, and are being tracked under CAP, to modify the 24" diameter MS system piping in the heater bay to eliminate the recurring pipe support deficiencies resulting from high-cycle fatigue loadings, and to reduce system vibration levels due to normal operating conditions of the system.

As part of ongoing corrective actions, the MS pipe supports are visually examined for discrepancies at the end of each operating cycle.

Attachment 2

Response to Request for Additional Information
for License Renewal Application
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) regarding the License Renewal Application is shown in italics. The Nebraska Public Power District's (NPPD) response to each RAI is shown in block font.

NRC Request: *RAI 3.5-1*

Background:
In the license renewal application (LRA) Section 3.5.2.2.1.1, Section 3.5.2.2.1.2, Section 3.5.2.2.1.4, Section 3.5.2.2.1.10, Section 3.5.2.2.2.1, Section 3.5.2.2.2.2, and Section 3.5.2.2.2.4, the applicant stated that there are no aging effects requiring management for Cooper Nuclear Station (CNS) concrete due to the following:

1. *Concrete is designed in accordance with ACI 318-63*
2. *The CNS below-grade environment is not aggressive*

Issue:
The staff is unable to verify the applicant claims due to lack of supporting data and/or information in the LRA in the following areas:

1. *Air-entrained value and water-cement ratio*
2. *Data for below-grade water chemistry*

Request:
In order to complete the staff's review, additional information is needed as follows:

A. *Provide supporting data/information for the above items.*
B. *Explain what action will be taken for inaccessible areas, if degradation (as a source of aging effects) is discovered in accessible areas? Will this involve additional commitments?*
    - *Explain why there are no aging effects requiring management for CNS concrete (both accessible and inaccessible) while the Generic Aging Lessons Learned (GALL) Report recommends the Structures Monitoring Program and/or a Plant-Specific Program to manage concrete aging effects for LRA Section 3.5.2.2.1.2, Section 3.5.2.2.2.1, and Section 3.5.2.2.2.4.*

NPPD Response:

Part A

The CNS concrete specification provided for air content between four and six percent. Concrete strength was established based on Method 2 of ACI 318-63. Method 2 provided for tests of trial mixes to ensure required concrete strength at water–cement ratios that provided sufficient workability. The maximum permissible water-cement ratio for the concrete used at CNS was that established by the water-cement ratio versus concrete strength curve produced by Method 2 that yielded an average strength which satisfied the requirements of ACI 318-63 Section 504 "Strength Test of Concrete." The maximum permissible water-cement ratio was 0.71 for concrete with 3000 psi strength and 0.52 for concrete with 4000 psi strength.

Sampling of three test wells in November 2006 yielded the following results for CNS below-grade water chemistry.

	Test Hole B-31	Test Hole B-12	Test Hole B-1
pH	7.0	8.0	7.6
Chloride	24.4 ppm	17.0 ppm	22.0 ppm
Sulfate	82.5 ppm	33.0 ppm	74.1 ppm

Part B

The CNS Structures Monitoring Program provides for inspections of accessible areas. If findings on accessible structures or components indicate that potential degradation may be occurring in inaccessible areas, an evaluation will be performed and appropriate corrective actions will be taken under the Corrective Action Program. This involves no additional commitments since the corrective action and confirmation processes are elements of the existing Structures Monitoring Program.

LRA Section 3.5.2.2.1.2 corresponds to NUREG-1800 Section 3.5.2.2.1.2 which discusses cracks and distortion due to increased stress levels from settlement, and reduction of foundation strength, cracking and differential settlement due to erosion of porous concrete subfoundations. As indicated in LRA Table 3.5.1 (Structures and Component Supports, NUREG-1801 Vol. 1), Item 3, the CNS containment is a Mark I steel containment with a foundation integral to the floor of the reactor building. The containment foundation is not exposed to a soil environment. Consequently, the NUREG-1801 items that reference this Standard Review Plan (SRP) section are not applicable to Mark I steel containments. The Structures Monitoring Program includes inspections of the reactor building structures to confirm the absence of aging effects caused by settlement. A porous concrete subfoundation is not a design feature of the CNS reactor building.

LRA Section 3.5.2.2.2.1 corresponds to SRP Section 3.5.2.2.2.1. Each discussion in LRA Section 3.5.2.2.2.1 for a specific aging effect is intended to explain why there is no aging effect requiring management based on the criteria provided in the SRP discussions and the associated criteria of NUREG-1801 Volume 2. Nevertheless, as indicated in LRA Chapter 3.5, the Structures Monitoring Program is applied to the affected structures to confirm the absence of significant aging effects. Accessible concrete inspected under the Structures Monitoring Program provides indication of the condition of inaccessible concrete since it is constructed to the same standards and is exposed to similar or more severe environments. For example, accessible exterior concrete is exposed to greater extremes of temperature than inaccessible concrete. For sliding surfaces addressed in Paragraph 8 of LRA Section 3.5.2.2.2.1, the Structures Monitoring Program and Inservice Inspection – IWF Program confirm the absence of aging effects requiring management.

LRA Section 3.5.2.2.2.4 corresponds to SRP Section 3.5.2.2.2.4. Each discussion in LRA Section 3.5.2.2.2.4 for specific aging effects is intended to explain why there are no aging effects requiring management based on the criteria provided in the SRP discussions and the associated criteria of NUREG-1801 Volume 2. Nevertheless, as indicated in LRA Chapter 3.5, the Structures Monitoring Program is applied to the affected structures to confirm the absence of significant aging effects. Per LRA Section B.1.36, Structures Monitoring Program activities will be enhanced to include examination of the exposed portions of the below grade concrete, when excavated for any reason. The program enhancements also include periodic monitoring of below-grade water chemistry to confirm that groundwater remains nonaggressive.

<u>NRC Request</u>: *RAI 3.5.2.2.2.2-1*

<u>Background</u>:
Standard Review Plan (SRP) Section 3.5.2.2.2.2, "Aging Management of Inaccessible Areas," consists of five sub-sections to review as follows:

1. *Loss of Material (spalling, scalling) and cracking due to freeze-thaw in below grade inaccessible concrete areas for group 1-3, 5, and 7-9 structures.*
2. *Cracking due to expansion and reaction with aggregates could occur in below-grade inaccessible concrete areas for group 1-5, and 7-9 structures.*
3. *Cracks and distortion due to increased stress levels from settlement and reduction of foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundation could occur in below-grade inaccessible concrete areas of groups 1-3, 5 and 7-9 structures.*
4. *Increase in porosity and permeability, cracking, loss of material due to aggressive chemical attack.*
5. *Increase in porosity and permeability, and loss of strength due to leaching of calcium hydroxide.*

Issue:
LRA Section 3.5.2.2.2.2 states that groups 1-3, 5 and 7-9 inaccessible concrete areas provided in accordance with specification ACI 318-63, Building Code Requirements for Reinforced Concrete and that concrete also meets requirements of later ACI guide 201.2R-77. The LRA further states that inspection of accessible concrete have not revealed degradation related to corrosion of embedded steel and that the below-grade environment is not aggressive. The LRA concludes that corrosion of embedded steel is not an aging effect requiring management for concrete (Reference to RAI 3.5-1). However, the staff was unable to complete its review because the LRA did not contain the related information for Sub-section 3.5.2.2.2.2.1 through Sub-section 3.5.2.2.2.2.5 of the SRP.

Request:
Provide the related information for Sub-section 3.5.2.2.2.2.1 through Sub-section 3.5.2.2.2.2.5 for the staff to review.

NPPD Response:

The related information in SRP Section 3.5.2.2.2.2 is addressed in LRA Section 3.5.2.2.2. Specifically, Item 1 is addressed by Section 3.5.2.2.2.1.4, Item 2 by Section 3.5.2.2.2.1.5, Item 3 by Section 3.5.2.2.2.1.6 and Section 3.5.2.2.2.1.7, and Item 4 by Section 3.5.2.2.2.1.2. The related information applicable to Item 5 (SRP Subsection 3.5.2.2.2.2.5) is provided in LRA Subsection 3.5.2.2.2.4.3. While LRA Subsection 3.5.2.2.2.4.3 is under the category of Group 6 structures, the discussion also applies to concrete in Groups 1-3, 5 and 7-9.

NRC Request: *RAI 3.5.2.2-1*

Background:
NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 1, states that the IWE Inservice Inspection Program should be supplemented and that additional appropriate examinations to detect stress-corrosion cracking (SCC) in bellows assemblies and dissimilar metal welds are warranted. In addition, Information Notice 92-20 describes instances of containment bellows cracking.

Issue:
LRA Section 3.5.2.2.1.7 states that the existing Containment Leak Rate and Containment Inservice Inspection – IWE Programs are adequate to detect cracking since the susceptible components are not subject to a corrosive environment. The staff could not determine the basis for this position.

Request:
The staff requests that the applicant provide a discussion on the augmented exams discussed in LRA Table 3.5-1, Items 10 and 11, and indicate how the exams will detect fine cracks.

NPPD Response:

The CNS Containment Inservice Inspection Program contains provisions for performing augmented examinations of components likely to experience accelerated degradation and aging. Components subject to augmented examination in accordance with ASME Section XI, Subsection IWE, Item 4.11 receive a VT-1 examination of the accessible areas based on the limitations set forth in 10 CFR 50.55a(b)(2)(ix)(G). Direct VT-1 visual examinations are conducted within a maximum examination distance of 2' (24") or less from the surface to be examined and with a minimum illumination level that is adequate to resolve the VT-1 sized characters (0.044" maximum height) on a character card. Examinations performed to this standard are capable of detecting fine cracks.

Although the CNS Inservice Inspection Program has no components subject to augmented examination in accordance with ASME Section XI, Subsection IWE Item 4.12, any components subject to this provision in the future would receive ultrasonic thickness examinations for inaccessible areas that cannot be examined visually. Examinations performed to this standard are similarly capable of detecting fine cracks.

The augmented examinations of the Containment Inservice Inspection Program discussed above are not applied to stainless steel components and dissimilar metal welds identified in LRA Table 3.5.1, Items 10 and 11 since those components have not been judged likely to experience accelerated degradation and aging. Information Notice 92-20 primarily entailed problems with local leak rate testing and did not identify a cause of the containment bellows cracking that would indicate similar CNS components are susceptible to accelerated degradation or aging. As stated in LRA Section 3.5.2.2.1.7, these components are not subject to a corrosive environment, which is a necessary factor for establishing susceptibility to SCC.

Within the Primary Containment Leak Rate Program, CNS will continue to perform a periodic integrated leak rate test of the overall primary containment, which is capable of detecting leakage in the unlikely event of through-wall cracking. Additionally, this program requires local leakage rate tests of expansion bellows.

In summary, the combination of the Containment Leak Rate and Containment Inservice Inspection Programs provides reasonable assurance that cracking will be managed.

NRC Request: *RAI 3.5.2.2-2*

Background:
NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 1, states that the IWE Inservice Inspection Program should be supplemented in that VT-3 visual inspection may not detect fine cracks.

Issue:
LRA Section 3.5.2.2.1.8 states that the existing containment Leak Rate Program with augmented exams and Containment Inservice Inspection – IWE will be used to detect cracking.
LRA Section 3.5.2.2.1.8 and LRA Table 3.5.1, Items 12 and 13 refer to augmented inspections but do not provide a discussion on the specifics of the augmented inspections.

Request:
The staff requests that the applicant provide a discussion on the augmented inspections, and indicate how they will be used to detect fine cracks.

NPPD Response:

LRA Table 3.5.1 Items 12 and 13 correspond to the analogous items in NUREG 1800, Table 3.5-1, "Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report." Upon further review, Items 12 and 13 of this SRP table were found not applicable for CNS components. As stated in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report" Items II.B4-3 (C-14) and II.B1.1-3 (C-20), the aging effect "cracking/cyclic loading" applies to these structural components only if a current licensing basis fatigue analysis does not exist. CNS does have fatigue analyses for the subject components. Accordingly, LRA Section 3.5.2.2.1.8 and Tables 3.5.1 and 3.5.2-1 have been revised (see Attachment 3, Changes 1, 2, 3, and 4).

NRC Request: *RAI 3.5.2.2.2.6-1*

Background:
NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 1, Generic Item T-30, recommends the Structures Monitoring Program to manage loss of material and general corrosion.

Issue:
LRA Section 3.5.2.2.2.6.2 states that the Structures Monitoring Program manages loss of material for steel structural components. The LRA further states that for some components the Fire Protection Program supplements the Structures Monitoring Program and, for other components, the Periodic Surveillance and Preventive Maintenance, Fire Protection, or Fire Water System Programs be used to manage the loss of material. The staff could not determine how the above mentioned programs meet or exceed the Structures Monitoring Program.

Request:
The staff requests that the applicant provide a comparison of the above mentioned programs to the Structures Monitoring Program and specify how the programs will meet or exceed the requirements of the Structures Monitoring Program, relative to the aging effect "loss of material/general and pitting corrosion."

NPPD Response:

A comparison of the three programs demonstrates that the Fire Protection, Fire Water System, and Periodic Surveillance and Preventive Maintenance Programs are as effective as the Structures Monitoring Program for managing the aging effect "loss of material/general and pitting corrosion."

For steel structural components, the Structures Monitoring Program manages loss of material through visual inspection of component surface condition at a frequency of at least once every five years. The acceptance criteria include no indications of loss of material, such as excessive rust or corrosion. The identified supplemental programs also employ the same visual inspection method, comparable frequencies, and the same acceptance criteria for managing loss of material for steel structural components. The parameter monitored for loss of material in all four programs is the condition of component external surfaces.

NRC Request: *RAI 3.5.2.2.2.6-2*

Background:
NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 1, Generic Item T-29, includes grouted anchors and grout as components/material which should be managed by the Structures Monitoring Program.

Issue:
LRA Section 3.5.2.2.2.6.1 states that the Structures Monitoring Program will confirm the absence of aging effects for CNS concrete components, but does not discuss grouted anchors.

Request:
The staff requests that the applicant discuss whether grout and grouted supports are included within the Structures Monitoring Program. In addition, if grout and grouted supports are not included within the Structures Monitoring Program, provide a discussion on how aging effects will be managed.

NPPD Response:

Grout pads for structural base plates are sub-components of the concrete group and not specifically identified as separate components. Therefore, structural grout (e.g. support base plates) is grouped with concrete. Per LRA Table 3.5.1 (Structures and Component Supports, NUREG-1801 Vol. 1), Item Number 40, grout pads for equipment are included with concrete components. Consequently, in LRA Table 2.4.4 (Bulk Commodities Subject to Aging Management Review) and Table 3.5.2-4 (Bulk Commodities Summary of Aging Management Evaluation), grout pads for equipment and support base plates are included with the concrete bulk commodity "Equipment pads/foundations." Accordingly, grout and grouted supports are included in the scope of the Structures Monitoring Program.

NRC Request: *RAI 3.5.2.2.2.6-3*

Background:
LRA Section 3.5.2.2.2.6.3 states that the CNS aging management review did not identify any component support structure/aging effect combination corresponding to NUREG-1801 Volume 2 Item III.B4.2-a.

Issue:
The staff could not determine whether vibration isolation elements exist at CNS and are included within the scope of license renewal, or whether the applicant has determined that vibration isolation elements included within the scope of license renewal have no aging effect.

Request:
The staff requests that the applicant indicate whether or not vibration isolation elements are included within the scope of license renewal at CNS. If vibration isolation elements are included within scope the scope of license renewal, provide a basis for why they are not covered by the Structures Monitoring Program.

NPPD Response:

Vibration isolation elements are not uniquely identified in the application because they are an integral part of the overall structural support component. These components are addressed in the LRA in Table 2.4-4 (Bulk Commodities Subject to Aging Management Review) and Table 3.5.2-4 (Bulk Commodities) under line item "Component and piping supports." Aging management activities for managing the entire structural support assembly under the Structures Monitoring Program will manage the effects of aging on these vibration isolation elements.

NRC Request: *RAI 3.5.2.3-1*

Background:
GALL Report," Revision 1, Generic Item TP-6, says stainless steel support members and bolted connections in an outdoor environment should be monitored by the Structures Monitoring Program for the aging effect loss of material.

Issue:
LRA Table 3.5.2-4 lists five stainless steel support and bolting component groups in an "air-outdoor" environment. The LRA lists the aging effect and aging management program as none and refers to Note I and Note 503, which state that aging management is not required for stainless steel components exposed to the external environment because the environment at CNS

is not chemically polluted. The staff did not determine that Note 503 is adequate to conclude that aging management is not required for these components.

Request:
Provide a basis for the conclusion that no aging effect is applicable to the above mentioned component groups and why the component groups were not included in the Structures Monitoring Program.

NPPD Response:

CNS external ambient environment is non-aggressive and does not include vapors of sulfur dioxide or sodium chloride from saltwater spray which could result in loss of material. Industry experience has shown that stainless steel is very resistant to general corrosion for both interior and exterior exposures and under conditions of high or low humidity. Stainless steel also contains chromium at levels sufficient to provide adequate resistance to corrosion in both industrial and marine environments. CNS is located in a predominantly agricultural environment not located near seawater, and no significant industrial plants are in the area which could result in an aggressive air-outdoor environment. Therefore, the conditions required to cause loss of material for stainless steel elements exposed to an air-outdoor environment is not present at CNS and loss of material is not an aging effect requiring management. Nevertheless, the Structures Monitoring Program includes inspections of the five stainless steel support and bolting component groups in an "air-outdoor" environment, with the exception of ASME Class 1, 2, 3 and MC supports bolting, which is included in the Inservice Inspection - IWF Program.

Attachment 3

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

This attachment provides changes to the License Renewal Application based on the responses to the Requests for Additional Information provided in Attachments 1 and 2, as well as for other clarifications. The changes are presented in underline/strikeout format.

1. LRA Section 3.5.2.2.8 is revised to read:

   "~~Cyclic loading can lead to cracking of steel and stainless steel penetration bellows, and dissimilar metal welds of BWR containments and BWR suppression pool shell and downcomers.~~

   ~~With proper design, cracking due to cyclic loading is not expected to occur in the drywell, torus and associated penetration bellows, penetration sleeves, unbraced downcomers, and dissimilar metal welds. A review of plant operating experience did not identify any cracking of these components, and primary containment leakage has not been identified as a concern. Nonetheless, the existing Containment Leak Rate Program with augmented exams and Containment Inservice Inspection – IWE will continue to be used to detect cracking. Observed conditions that have the potential for impacting an intended function are evaluated or corrected in accordance with the corrective action process. The Containment Inservice Inspection – IWE and Containment Leak Rate programs are described in Appendix B.~~

   In accordance with NUREG-1801 Volume 2, Items II.B1.1-3 and II.B4-3, this line is applicable only for structures without a CLB fatigue analysis. CNS has a CLB fatigue analysis for these components."

   Reference: Response to RAI 3.5.2.2-2.

2. The Discussion column entry for LRA Table 3.5.1 (Structures and Component Supports, NUREG-1801 Vol. 1), Item Number 3.5.1-12, Page 3.5-24 is revised to read:

   "~~Consistent with NUREG-1801. With proper design, cracking due to cyclic loading is not expected to occur. Nonetheless, the Containment Leak Rate Program with augmented exams and Containment Inservice Inspection will continue to be used to detect cracking. The Containment Inservice Inspection Program includes augmented exams to detect fine cracks.~~ Not applicable. CNS has a CLB fatigue analysis for these components.

   See Section 3.5.2.2.1.8."

Reference: Response to RAI 3.5.2.2-2.

3. The Discussion column entry for LRA Table 3.5.1 (Structures and Component Supports, NUREG-1801 Vol. 1), Item Number 3.5.1-13, Page 3.5-25, is revised to read:

"~~With proper design, cracking due to cyclic loading is not expected to occur. Nonetheless, the Containment Leak Rate Program with augmented exams and Containment Inservice Inspection will continue to be used to detect cracking. The Containment Inservice Inspection Program includes augmented ultrasonic exams to detect fine cracks.~~ Not applicable. CNS has a CLB fatigue analysis for these components.

See Section 3.5.2.2.1.8."

Reference: Response to RAI 3.5.2.2-2.

4. Table 3.5.2-1 (Reactor Building and Primary Containment Summary of Aging Management Evaluation), Page 3.5-58, is revised to read:

Primary containment mechanical penetrations (includes those with bellows)	PB,SSR	Carbon steel	Air – indoor uncontrolled	Cracking	~~CII-IWE Containment Leak Rate~~ TLAA – metal fatigue	~~II.B4-3 (C-14)~~ II.B4-4 (C-13)	~~3.5.1-12~~ 3.5.1-9	~~B~~ A

Reference: Response to RAI 3.5.2.2-2.

5. Section B.1.18 (Flow Accelerated Corrosion), Exceptions to NUREG-1801, Page B-57 is revised to read:

"**Exceptions to NUREG-1801**

The FAC Program is consistent with the program described in NUREG-1801, Section XI.M17, Flow-Accelerated Corrosion, with the following exception.

Elements Affected	Exception
4. Detection of Aging Effects	NUREG-1801 recommends using both ultrasonic (UT) and radiographic testing to detect wall thinning. CNS uses UT for detecting wall thinning~~only~~. [1]

Exception Note

1. ~~This is sufficient because, a~~As stated in NSAC-202L Revision 2, both UT and RT methods can be used to investigate whether or not wear is present. However, the UT method provides more complete data for measuring the remaining wall thickness. As a result UT is the preferred method for detecting wall thinning. The CNS program does not preclude the use of other inspection techniques such as RT if conditions do not permit the use of UT."

Reference: Clarification based on discussions with Nuclear Regulatory Commission inspectors during License Renewal Regional Inspection.