Entergy

Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

Michael Perito
Vice President, Operations
Grand Gulf Nuclear Station
Tel. (601) 437-6409

GNRO-2012/00080

July 25, 2012

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Response to Request for Additional Information (RAI) Set 24 dated June 27, 2012
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCE: NRC Letter, "Requests for Additional Information for the Review of the Grand Gulf Nuclear Station, License Renewal Application," dated June 27, 2012 (GNRI-2012/00142)

Dear Sir or Madam:

Entergy Operations, Inc. is providing, in the Attachment, the response to the referenced Request for Additional Information (RAI).

This letter contains no new commitments. If you have any questions or require additional information, please contact Christina L. Perino at 601-437-6299.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 25th day of July, 2012.

Sincerely,

MP/JAS

Attachment: Response to Request for Additional Information (RAI)

cc: with Attachment

Mr. John P. Boska, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8-C2
Washington, DC 20555

cc: without Attachment

Mr. Elmo E. Collins, Jr.
Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
1600 East Lamar Boulevard
Arlington, TX 76011-4511

U.S. Nuclear Regulatory Commission
ATTN: Mr. A. Wang, NRR/DORL
Mail Stop OWFN/8 G14
11555 Rockville Pike
Rockville, MD 20852-2378

U.S. Nuclear Regulatory Commission
ATTN: Mr. Nathaniel Ferrer NRR/DLR
Mail Stop OWFN/ 11 F1
11555 Rockville Pike
Rockville, MD 20852-2378

NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

Attachment to

GNRO-2012/00080

Response to Request for Additional Information (RAI)

RAI 3.3.2.15-1

Background. License renewal application (LRA)Tables 3.3.2-15, "Standby Diesel Generator System," and 3.3.2-16, "HPCS Diesel Generator System," contain aging management review (AMR) items which specify the Service Water Integrity program to manage loss of material due to wear for copper alloy with greater than 15 percent zinc heat exchanger tubes externally exposed to treated water. The AMR items cite generic note H indicating that this aging effect is not in the Generic Aging Lessons Learned (GALL) Report for this component, material, and environment combination.

LRA Section B.2.1.14 states that the applicant's Service Water Integrity Program is consistent with GALL Report aging management program (AMP) XI.M20, "Open-Cycle Cooling Water System." GALL Report AMP XI.M20 manages aging effects on components exposed to raw water systems using a combination of (a) surveillance and control techniques, (b) inspections, and (c) heat transfer capability testing of heat exchangers. The GALL Report AMP XI.M20 "detection of aging effects" program element states that visual or nondestructive examinations may be performed to identify degradation and that nondestructive testing, such as ultrasonic testing and eddy current testing, are effective methods to measure surface conditions or the extent of wall loss associated with service water system piping and components.

Issue. The GALL Report identifies loss of material due to various types of corrosion as an aging effect requiring management for copper alloy heat exchanger components; but does not identify loss of material due to wear as an aging effect requirement management. The nature of the loss of material due to wear aging mechanism for these items is not clear to the staff.

The applicant proposes to manage the external surface of heat exchange tubes exposed to treated water using a program that is intended to manage aging for components exposed to raw water. While it is possible to detect loss of material on the external portion of heat exchanger tubes by performing certain nondestructive examinations of the internal surfaces, the LRA did not describe the nondestructive examination method that will be used to manage the aging effects for these items.

Request. For the items in Tables 3.3.2-15, and 3.3.2-16 that are being managed for loss of material due to wear, describe the nature of the aging mechanism and discuss the associated plant-specific or industry operating experience associated with the identification of the additional aging mechanism for these items. Also, provide a description of the nondestructive examination method to be used by the Service Water Integrity Program and justification that the method being used will adequately manage loss of material from the treated water side of the associated heat exchanger tubes.

RAI 3.3.2.15-1 RESPONSE

For the items in Tables 3.3.2-15, and 3.3.2-16, wear could occur on the external side of the tubes of the heat exchanger that may experience intermittent relative motion between the tubes and the tube support members of the heat exchanger. Related to operating experience, loss of material due to wear on heat exchanger tubes is evaluated for heat exchangers that experience very low run times. Wear, if applicable, is a design condition that would show up early in life on heat exchangers in continuous service and would have likely been corrected such that it is not an issue for license renewal. Diesel generator heat exchangers, such as these, normally have

less than twenty five hundred hours of service after forty years and may be subject to this aging effect.

The Service Water Integrity Program implements and maintains an ongoing program of surveillance for heat exchangers subject to loss of material due to wear by periodically performing visual inspections and non destructive examination of tubes by eddy current testing, which is capable of locating wall thinning at specific locations on the external surfaces of the tubes. These periodic and ongoing inspections assure that loss of material due to wear on the treated water side of the tubes is adequately managed.

RAI 3.5.1.87-1

Background. The GALL Report recommends that structural bolting in any environment should be managed for loss of preload due to self-loosening (e.g., GALL Report items III.B1.1.TP-229 and III.A1.TP-261). Standard Review Plan for License Renewal (SRP-LR) Table 3.5-1, items 87 and 88, recommends the ASME Section XI, Subsection IWF and Structures Monitoring Program, respectively, to manage this aging effect.

Issue. LRA Table 3.5.1, items 87 and 88, state that this aging effect is not applicable and does not require management. No further discussion is provided.

Request. Explain why loss of preload is not an aging effect for structural bolting within the scope of license renewal, or provide an acceptable AMP to manage loss of preload during the period of extended operation. The response should address both the ASME Section XI, Subsection IWF and the Structures Monitoring Programs.

RAI 3.5.1.87-1 RESPONSE

Vibration, flexing of the joint, cyclic shear loads, thermal cycles and other conditions can cause self-loosening of a fastener. These causes of loosening are minor contributors in structural steel and steel component threaded connections and are eliminated by initial preload bolt torquing. Grand Gulf Nuclear Station (GGNS) uses site procedures and manufacturer recommendations to provide guidance for proper torquing of nuts and bolts used in structural applications. Additionally, operating experience has not shown self-loosening of structural bolting used at GGNS. Therefore, loss of preload due to self-loosening is not an aging effect requiring management for structural steel and steel component threaded fasteners within the scope of license renewal. This includes structural bolting addressed under the ASME Section XI, Subsection IWF Program and under the Structures Monitoring Program.

RAI 3.5.2.1-1

Background. LRA Table 3.5.2-1 addresses elastomer containment building electrical seals and sealant that will be managed for the aging effects of cracking and change in material properties by the Containment Leak Rate Program and references SRP-LR Table 3.5.1, item 26. The GALL Report recommends using GALL Report AMP XI.S1, "ASME Section XI, Subsection IWE," to ensure that these aging effects are adequately managed.

Issue. In the LRA, the applicant references SRP-LR Table 3.5.1, item 26, and credits the Containment Leak Rate Program to manage the aging effect of cracking and change in material properties of elastomer containment building electrical seals and sealant in an air – indoor uncontrolled environment. Since the Containment Leak Rate Program is a performance monitoring program that monitors parameters related to leakage rates and does not by itself provide information that would indicate aging degradation has initiated, the staff is unable to determine how the Containment Leak Rate Program will evaluate change in material properties, or if visual examinations are performed of the elastomer containment building electrical penetration seals and sealants to manage aging in the form of cracking or change in material properties.

Request. Describe how the Containment Leakage Rate Program meets the recommendations in GALL Report item II.B4.CP-40 with respect to managing the aging effect of cracking and change in material properties of elastomer containment building electrical seals and sealant in an air – indoor uncontrolled environment.

RAI 3.5.2.1-1 RESPONSE

NUREG-1801 item II.B4.CP-40 applies to moisture barriers. The appropriate NUREG-1801 item for containment penetration seals and sealants is NUREG-1801 item II.B4.CP-41. To better align with the recommendations of NUREG-1801, the License renewal application (LRA) Table 3.5.2-1 line item for containment building electrical penetration seals and sealants is revised to reference NUREG-1801 item II.B4.CP-41. As a result of this change, the references to Table 1 Item 3.5.1-26 and Note "E" are revised to 3.5.1-33 and Note "A" as shown below. This change makes the line item consistent with NUREG-1801 recommendations for managing the aging effects of cracking and change in material properties on containment penetration seals and gaskets using the Containment Leak Rate Program.

Additions are shown with underline and deletions with strikethrough.

Table 3.5.2-1: Containment Building

Structure and/or Component or Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Containment building electrical penetration seals and sealant	PB, SSR	Elastomer	Air – indoor uncontrolled	Cracking Change in material properties	Containment Leak Rate	~~II.B4.CP-40~~ II.B4.CP-41	~~3.5.1-26~~ 3.5.1-33	~~E~~ A

RAI 3.5.2.4-4

Background. LRA Table 3.5.2-4 addresses carbon steel fire hose reels that will be managed for loss of material by the Fire Water System Program and references SRP-LR Table 3.5.1, item 3.5.1-92. The GALL Report recommends using GALL Report AMP XI.S6, "Structures Monitoring Program," to ensure that these aging effects are adequately managed.

Issue. The staff notes that the applicant's Fire Water System Program has been enhanced to include periodic inspection of fire hose reels; however, GALL Report AMP XI.M27, "Fire Water System," focuses on the inspection of internal system corrosion conditions. The staff is unclear how the applicant's Fire Water System Program will be utilized to address the structure/aging effect combination, during the period of extended operation.

Request. Describe how the Fire Water System Program meets the recommendations provided in GALL Report AMP XI.S6, with respect to frequency of inspections, qualifications of inspection personnel, and acceptance criteria.

RAI 3.5.2.4-4 RESPONSE

The Fire Water System aging management program NUREG -1801, Section XI.M27, Element 4, Detection of Aging Effects, addresses detection of corrosion by visual inspection of structural steel commodities (i.e. yard fire hydrants). LRA Section B.1.21, "Fire Water System" Element 4, includes an enhancement for visual inspection of hose reels which aligns with NUREG-1801, AMP XI.S6, "Structures Monitoring Program" (SMP) for detection of loss of material due to corrosion for steel components. The enhancement also includes revising the acceptance criteria to verify no unacceptable degradation which also aligns with the SMP. The frequency of inspections recommended in NUREG-1801, AMP XI.M27 is annually, which is more frequent than the inspection frequency identified for the SMP. The SMP recommends inspector qualifications consistent with industry guidelines and standards, and guidelines for implementing the requirements of 10 CFR 50.65. In accordance with GGNS procedures, personnel performing Fire Water System Program inspections must be qualified to perform those activities. The qualifications of the inspectors are consistent with industry guidelines and standards. The SMP also recommends that qualified engineering personnel evaluate inspection results based on acceptance criteria selected for each structure/aging effect to ensure that the need for corrective actions is identified before loss of intended functions. The Fire Water System Program directs personnel to initiate appropriate corrective actions when deficiencies are identified to ensure inspection results are evaluated by qualified engineering personnel based on the program acceptance criteria. As described, the inspection frequency, personnel qualification and acceptance criteria specified by the Fire Water System Program for inspection of carbon steel fire hose reels for loss of material align with the recommendations provided in the SMP.

RAI 3.5.2.4-5

Background. LRA Table 3.5.2-4 states that rubber water stops exposed to an air – indoor uncontrolled environment have no aging effects requiring management and no proposed AMP. GALL Report Section IX.F, "Selected Definitions & Use of Terms for Describing and Standardizing Aging Mechanisms," identifies elastomer (i.e., rubber) degradation mechanisms as cracking, crazing, fatigue breakdown, abrasion, chemical attacks, and weathering. The GALL Report states that for rubber materials, hardening and loss of strength of elastomers can be induced by elevated temperature (over about 95°F (35°C)), and additional aging factors such as exposure to ozone, oxidation, and radiation. The staff notes that GALL Report item AP-102 states that elastomers exposed to an air-indoor uncontrolled environment are subject to hardening and loss of strength due to degradation, and can be managed by GALL Report AMP XI.M36, "External Surfaces Monitoring of Mechanical Components." The staff also notes that SRP-LR Table 3.5-1, item 72, identifies loss of sealing due to deterioration of seals, gaskets,

and moisture barriers (caulking, flashing, and other sealants) as an aging effect/mechanism that will be managed by the Structures Monitoring Program.

Issue. The staff lacks sufficient information to determine whether these rubber water stops are exposed to sufficient levels of ozone, oxidation, and radiation to cause aging effects requiring management.

Request. Provide a technical basis as to why there are no aging effects requiring management or AMP for these rubber water stops or propose an AMP to manage the aging effects.

RAI 3.5.2.4-5 RESPONSE

The rubber water stops shown in LRA Table 3.5.2-4 are located in the expansion joint between the turbine building foundation basemat and adjoining structures, and the expansion joint between the turbine building foundation basemat and the turbine generator foundation. The rubber water stops are partially embedded in the concrete and the portion exposed to the air-indoor uncontrolled environment is located within the expansion joint gap. The rubber water stops are placed within the expansion joint a minimum of three feet below the top surface of the six foot thick turbine building foundation basemat. Additionally, the expansion joints containing the water stops are filled with an elastomeric material that is within the scope of license renewal. The expansion joint is identified in LRA Table 3.5.2-4 as component "seismic isolation joint." The Structures Monitoring Program manages the effects of aging on the elastomer material within the seismic isolation joint. The maximum area temperature for the interior of the turbine building is 105°F. Since the water stops are in the foundation basemat, the lowest elevation of the building, they are not exposed to temperatures above the NUREG-1801 Table IX.C threshold for elastomer degradation due to high temperature of about 95°F. With the protection provided by the expansion joint configuration, the water stop material is not exposed directly to an air-indoor uncontrolled environment. Exposure to an adverse environment (elevated temperatures, exposure to ozone, oxidation and radiation) for the water stops is not credible since the encapsulation of the water stop by the surrounding concrete and expansion joint filler material provides ample protection. Therefore, there are no aging effects requiring management for rubber water stops.

RAI 3.5.2.2.1.3-1

Background. SRP-LR Section 3.5.2.2.1.3 addresses loss of material due to general, pitting and crevice corrosion for steel elements of accessible and inaccessible areas of containments. The SRP-LR recommends further evaluation if the four following GALL Report conditions cannot be satisfied:

1. Concrete meeting the specifications of ACI 318 or 349 and the guidance of ACI 201.2R was used for the containment concrete in contact with the embedded containment shell or liner.
2. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner.
3. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with ASME Section XI, Subsection IWE requirements.
4. Water ponding on the containment concrete floor is not common and when detected is cleaned up in a timely manner.

Issue. The staff agrees that conditions 2 and 4 were addressed adequately by the applicant; however, the LRA did not specify that condition 1 was met or for condition 3 if a moisture barrier was present at the junction where the liner becomes embedded in the concrete and, if present, is it subject to aging management activities in accordance with ASME Section XI Subsection IWE requirements.

Request.

a. Confirm that the containment concrete in contact with the embedded steel liner meets the guidance contained in ACI 201.2R as specified in the GALL Report.

b. State whether a moisture barrier is or is not present at the junction where the liner becomes embedded in the concrete and, if present, that it is subject to aging management activities in accordance with ASME Section XI Subsection IWE requirements.

RAI 3.5.2.2.1.3-1 RESPONSE

a. Gulf Nuclear Station (GGNS) concrete structures were designed and constructed in accordance with American Concrete Institute (ACI) 318-71, *Building Code Requirements for Reinforced Concrete*, and ACI 349-72, *Criteria for Reinforced Concrete Nuclear Power Containment Structures* which meets many of the recommendations of ACI 201.2R, *Guide to Durable Concrete,* by providing low permeability concrete by utilizing a low water-cement ratio based on minimum slump. The original construction specification for GGNS containment structure requires 5000 psi strength concrete. ACI 318-71 provides the recommendations for selection of cement, aggregates, air-entraining admixture, water-cement ratio, and reinforcing bar to attain a workable, consistent concrete mix in order to produce a quality durable concrete structure with the required compressive strength. In line with the recommendations of ACI 201.2R, ACI-318 requires material selection and testing to conform to the applicable ASTM standards (e.g., concrete aggregates shall conform to "Specifications for concrete aggregates" ASTM C 33, Portland cement shall conform to "Specifications for Portland cement" ASTM C 150, and air-entraining admixtures shall conform to "Specifications for Air-Entraining Admixtures for Concrete" ASTM C260). It also provides concrete placement and curing of the concrete to ensure the required quality, durability, and strength of the concrete for each application. ACI 201.2R, which was published after construction of GGNS concrete structures, provides recommendations for concrete construction to address the potential corrosion of embedded steel which could result in concrete degradation. In summary, the use of the ASTM standards and compliance with ACI 318 ensures concrete quality in contact with embedded steel liner also meets guidance recommended in ACI 201.2R.

b. Grand Gulf Nuclear Station (GGNS) does not have a moisture barrier at the junction where the containment liner becomes embedded in the concrete. The liner plate is stainless steel at the lower elevations of the containment including where it is embedded in concrete.

RAI 4.6.2-1

Background. LRA Section 4.6.2 states that calculations were identified for the bellows on the containment penetration guard pipes and the fuel transfer tube that analyzed a large number of cycles of flexure due to normal operation and earthquakes. The LRA also states that the number of analyzed cycles is significantly higher than the total number of cycles projected for

the period of extended operation. The LRA further states that the applicant will use the Fatigue Monitoring Program to manage the aging effects due to fatigue in accordance with 10 CFR 54.21(1)(c)(iii).

Issue. The LRA does not provide additional information regarding the calculations performed for the fatigue analysis, such as the number of analyzed cycles or the included transients. The staff needs further information to confirm that an evaluation for the fatigue analysis is valid for the period of extended operation

Request. Identify the number of cycles of the bellows for the guard pipes and fuel transfer tube were designed for and explain how that number was developed. Also identify which transients were included in the original evaluation and which transients will be monitored under the Fatigue Monitoring Program.

RAI 4.6.2-1 RESPONSE

The design specification for the penetration bellows, including the fuel transfer tube bellows, identifies specific values of movement for normal conditions and for accident conditions. Those values are bounding values that are greater than the movement expected during the transients. The bellows manufacturer then determined an allowable number of cycles for that bounding movement. The allowable number of cycles for that movement is much larger than the combined total allowable number of all transients identified in LRA Table 4.3-1.
The main steam secondary containment penetration bellows at the turbine building to auxiliary building interface has the lowest number of allowable cycles for penetration bellows. These bellows have an allowable value of 10,000 cycles of movement. The number of times the main steam lines are heated up (the lines are heated up as part of plant startup) plus the total number of earthquake cycles will remain well below the allowable 10,000 cycles through the period of extended operation. Other penetration bellows are qualified for many more movement cycles (up to 10^5 or 10^6 cycles). The allowable cycles are significantly higher than the total number of piping thermal movements, pressurization cycles, and seismic movement cycles expected for the period of extended operation.

RAI B1.14-1

Background. LRA Section B.1.14, states that the Containment Inservice Inspection – IWL Program is consistent with GALL Report AMP XI.S2, "ASME Section XI, Subsection IWL." The GALL Report AMP XI.S2 "parameters monitored or inspected" program element recommends that the containment concrete surfaces be examined for evidence of damage or degradation, such as those defined in ACI 201.1R and ACI 349.3R.

Issue. Grand Gulf Nuclear Station (GGNS) License Renewal Project, Aging Management Program Evaluation Report for Civil/Structural for Containment Inservice Inspection (CII) – IWL, only identifies ACI 349.3R criteria for the concrete containment surface examination, and does not refer to ACI 201.1R, "Guide for Conducting a Visual Inspection of Concrete in Service," for conducting containment concrete surface visual examination.

Request. Describe the methods that will be used for conducting containment concrete visual surface examination. In addition, describe if these methods are consistent with the guidance provided in ACI 201.1R.

RAI B1.14-1 RESPONSE

Grand Gulf Nuclear Station (GGNS) Containment Inservice Inspection-IWL Program is consistent with the parameters monitored and inspected as outlined in American Society of Mechanical Engineers (ASME) Section XI-IWL 2500. GGNS performs a general visual examination either directly or remotely with sufficient illumination (natural or artificial) and resolution (suitable for the local environmental conditions) to assess the general condition of the accessible containment concrete surfaces from permanent vantage points. The objective of this examination is to detect evidence of degradation or distress of the concrete surfaces being examined which could affect the structural integrity of the containment. Remote techniques, when used, are demonstrated to detect a 1/32" black line on a neutral gray card. Alternatively, the responsible engineer may approve the use of other remote examination techniques subject to the provisions of 10 CFR 50.55a. Examiners shall visually examine the surface areas to be inspected and document the results of the inspection on appropriate forms identified in the procedure. American Concrete Institute (ACI) 201.1R states that personnel conducting the condition survey must select those items important to the specific concerns relating to the reasons for the survey. The conduct of these inspections is in accordance with ASME Section XI-IWL and the guidance of ACI 349.3R "Evaluation of Existing Nuclear Safety-Related Concrete Structures." ACI 201.1R "Guide for Making a Condition Survey of Concrete in Service" is used as input to Containment Inservice Inspection-IWL Program for establishing appropriate parameters to be monitored and inspected. Therefore, the inspection methods are consistent with the guidance in ACI 201.1R.

RAI B1.14-2

Background. LRA Section B.1.14, states that the Containment Inservice Inspection – IWL Program is consistent with GALL Report AMP XI.S2. The GALL Report AMP XI.S2 "acceptance criteria" program element recommends that quantitative acceptance criteria based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R-02 may be used to augment the qualitative assessment of the Responsible Engineer.

Issue. The applicant's Containment Inservice Inspection (CISI) Program Plan provides examination process of IWL Program in Appendix B – "CII Examination Process Flowchart" and Appendix C – "GGNS Recording and Screening Criteria" in the applicant's procedure of CEP-CII-004, Rev. 302 "General and Detailed Visual Examinations of Concrete Containments." It is not clear if the quantitative acceptance criterion for containment concrete surface visual examination is included in the applicant's program.

Request. Describe the acceptance criteria used for the containment concrete surface visual examination and describe if the acceptance criteria are consistent with the quantitative acceptance criteria recommended in Chapter 5 of the ACI 349.3R-02.

RAI B1.14-2 RESPONSE

The Grand Gulf Nuclear Station (GGNS) acceptance criteria used for the containment concrete surface visual examination state, if any of the conditions listed below are present, the condition must be recorded on the examination form and the form forwarded to the appropriate personnel for acceptance review.

1. Active leaching or chemical attack to include areas of exudation, efflorescence, stalactites or stalagmites
2. Active abrasion or erosion degradation
3. Popouts or voids 50 mm (2 inches) or more in diameter (or of equivalent surface area)
4. Scaling 30 mm (1 - 1/8 inches) or more in depth
5. Spalling 20 mm (3/4 inches) or more in depth
6. Spalling 200mm (8 inches) or more in any dimension
7. Excessive corrosion of embedded metallic surfaces
8. Corrosion staining from corrosion of reinforcing steel or from an unknown source on the concrete surface
9. Cracks 1 mm (0.04 inches) in maximum width, measured below any surface enhanced widening
10. Excessive deflection, settlement or other physical movement
11. Conditions that indicate the presence of or cause degradation of inaccessible concrete

In addition, indications listed above are considered active unless the indications have been previously reported and remain essentially unchanged, or the evaluated indications have been determined inactive. These acceptance criteria are consistent with the quantitative acceptance criteria recommended in Chapter 5 of ACI 349.3R-02.

RAI B1.14-3

Background. LRA Section B.1.14, states that the Containment Inservice Inspection – IWL Program is consistent with the GALL Report AMP XI.S2. The GALL Report AMP XI.S2 "scope of program" program element recommends that scope of the AMP should be in accordance with ASME Section XI, Subsection IWL-1000. According to IWL-1100, steel embedded plates that are backed by concrete are within the scope of the containment concrete inservice inspection program. In addition, the GALL Report AMP XI.S2 "operating experience" program element states that the implementation of Subsection IWL, in accordance with 10 CFR 50.55a, is a necessary element of aging management for concrete containments through the period of extended operation.

Issue. During the walkdown of GGNS containment structure on January 31, 2012, the staff noted embedded steel plates in the concrete containment's exterior surface. These embedded steel plates had signs of corrosion and the concrete surface adjacent to these embedded plates had rust stains.

Requests. Please describe how the aging of the embedded steel plates located on the exterior surface of the concrete containment will be managed.

RAI B1.14-3 RESPONSE

The aging effect of the embedded steel plates located on the exterior surface of the concrete containment will be managed under the Containment Inservice Inspection – IWL (CII-IWL) Program in accordance with ASME Section XI IWL-1100. The containment dome surface area was evaluated during the last inspection interval and the inspection results indicated that the corrosion on the embedded plates did not exceed the screening criteria and would not jeopardize the structural integrity or leak tightness of the containment. The conditions were found acceptable without the need for engineering evaluation or other corrective action. These embeds are included as part of line item "Containment cylinder wall and dome" in LRA Table 3.5.2-1.