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GNRO-2012/00092

August 15, 2012

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

SUBJECT: Response to Requests for Additional Information (RAI) Set 28 dated July 23, 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 July 23, 2012 (GNRI-2012/00156) (ML12188A516)

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 15th day of August, 2012.

Sincerely,

MP/jas

Attachment: Response to Requests for Additional Information (RAI)

cc: (see next page)

cc: with Attachment

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Attachment to

GNRO-2012/00092

Response to Requests for Additional Information (RAI)

The format for the RAI responses below is as follows. The Request for Additional Information (RAI) is listed in its entirety as received from the Nuclear Regulatory Commission (NRC) with background, issue and request subparts. This is followed by the Grand Gulf Nuclear Station (GGNS) RAI response to the individual question.

RAI B.1.8-2a

Background. By letter dated May 1, 2012, Entergy Operations, Inc. (the applicant) responded to request for additional information (RAI) B.1.8-2, which requested that the license renewal application (LRA) include specific references to the Boiling Water Reactor and Vessels Internal Project (BWRVIP) documents credited for the applicant's Boiling Water Reactor (BWR) Penetrations Program. In its response, the applicant stated that its BWR Penetrations Program is consistent with the program described in NUREG-1801, Section XI.M8, BWR Penetrations, without exception. Therefore, by reference, the BWR Penetrations Program incorporates the relevant staff-approved BWRVIP documents consistent with NUREG-1801 guidance.

Issue. 10 CFR 54.21(d) requires that the updated final safety analysis report (UFSAR) supplement contain a summary description of the programs and activities for managing the effects of aging. Without referencing specific BWRVIP documents credited for the BWR Penetrations Program, the staff cannot determine whether the proposed UFSAR supplement in LRA Section A.1.8 contains an adequate summary description of the program and activities for managing the effects of aging in accordance with 10 CFR 54.21(d).

Request. Justify why LRA Section A.1.8 (UFSAR supplement) does not identify specific references to the BWRVIP documents credited for the BWR Penetrations Program.

RAI B.1.8-2a RESPONSE

LRA Section A.1.8 is revised to include a reference to BWRVIP-27-A, BWRVIP-47-A, and BWRVIP-49-A as shown below with additions underlined.

A.1.8 BWR Penetrations Program

The BWR Penetrations Program manages cracking of BWR vessel penetrations using inspection and flaw evaluation activities. Applicable industry standards and staff-approved BWRVIP documents including BWRVIP-27-A, BWRVIP-47-A, and BWRVIP-49-A are used to delineate the program.

LRA Section A.1.8 is revised as shown below with additions underlined

A.1.8 BWR PENETRATIONS

The BWR Penetrations Program manages cracking of BWR vessel penetrations using inspection and flaw evaluation activities. Applicable industry standards and staff-approved BWRVIP documents including BWRVIP-27-A, BWRVIP-47-A, and BWRVIP-49-A are used to delineate the program.

RAI B.1.8-4

Background. LRA Table 3.1.2-1 addresses the applicant's aging management for the reactor vessel, including reactor vessel nozzles and penetrations. LRA Table 3.1.2-1 indicates that the BWR Vessel Internals and Water Chemistry Programs manage cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC) and cyclic loading of the control rod drive (CRD) housing penetrations and incore housing penetrations under LRA items 3.1.1-102 and 3.1.1-98, respectively.

In contrast, the program scope of the BWR Penetrations Program and Generic Aging Lessons Learned (GALL) Report item IV.A1.RP-369 recommend that the BWR Penetrations Program and Water Chemistry Program be used to manage cracking of the CRD housing and incore monitor housing penetrations.

Issue. The applicant's aging management review results for cracking of the CRD housing and incore housing penetrations are not consistent with the scope of the BWR Penetrations Program and GALL Report item IV.A1.RP-369.

Request. Justify why cracking due to SCC, IGSCC and cyclic loading of the CRD housing penetrations and incore housing penetrations is managed by the BWR Vessel Internals and Water Chemistry Programs, which is inconsistent with the scope of GALL Aging Management Program (AMP) XI.M8, "BWR Penetrations," and GALL AMP item IV.A1.RP-369.

RAI B.1.8-4 RESPONSE

Inspection guidance for the control rod drive housings and incore housings is provided in BWRVIP-47-A, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines. For the purposes of comparisons to NUREG-1801, the BWR Penetrations Program described in XI.M8, which references BWRVIP-47-A, better represents the aging management program requirements.

LRA Table 3.1.2-1 lines for cracking of the control rod drive and incore housings are revised to identify the BWR Penetrations and Water Chemistry Control – BWR Programs. Additions are marked with underline and deletions are marked with strikethrough.

Table 3.1.2-1:	**Reactor Vessel**							
Component Type	**Intended Function**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801 Item**	**Table 1 Item**	**Notes**
CRD housings	Pressure boundary	Stainless steel, nickel alloy	Treated water > 140°F (int)	Cracking	BWR Penetrations ~~Vessel Internals~~ Water Chemistry Control – BWR	IV.A1.RP-369 ~~B1.R-104~~	3.1.1-98 ~~102~~	A ~~C~~
Incore housings	Pressure boundary	Stainless steel, nickel alloy	Treated water > 140°F (int)	Cracking	BWR Penetrations ~~Vessel Internals~~ Water Chemistry Control – BWR	IV.A1.RP-369	3.1.1-98	A ~~E~~

LRA Table 3.1.1, Item 3.1.1-98 is revised accordingly. Deletions are marked with strikethrough.

Table 3.1.1: Reactor Coolant System					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
3.1.1-98	Stainless steel or nickel alloy penetrations: instrumentation and standby liquid control exposed to reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, cyclic loading	Chapter XI.M8, "BWR Penetrations," and Chapter XI.M2, "Water Chemistry"	No	Cracking in stainless steel and nickel alloy nozzles and penetrations in the reactor vessel is managed by the Water Chemistry Control – BWR Program and ~~either~~ the BWR Penetrations ~~or BWR Vessel Internals~~ Program.

RAI B.1.9-1a

Background. By letter dated May 1, 2012, the applicant responded to RAI B.1.9-1 to address the scope of the BWR Stress Corrosion Cracking Program. In its response, the applicant clarified that the program applies to BWR piping and piping welds made of austenitic stainless steel and nickel alloy that is 4 inches or larger in nominal diameter and contains reactor coolant at a temperature above 93 °C (200 °F) during power operation regardless of code classification, consistent with the GALL Report. The staff notes that the scope of GALL Report, AMP XI.M7, "BWR Stress Corrosion Cracking," includes relevant piping and piping welds regardless of code classification (i.e., the GALL Report coverage of components is more inclusive than the ASME Code Class 1 components).

LRA Section 2.3.1.2 addresses the reactor coolant pressure boundary (RCPB) and indicates that the RCPB corresponds to American Society of Mechanical Engineers (ASME) Code Class 1 components. For example, LRA Section 2.3.1.2 states that the majority of the components that comprise the RCPB are from the nuclear boiler system and reactor recirculation system and the RCPB review also includes the Class 1 portions of various systems connected to the reactor vessel.

Issue. LRA Sections B.1.9 (program description) and Section A.1.9 (UFSAR supplement) references the RCPB as the scope of the program, rather than the recommended GALL Report's description of "relevant piping and piping welds regardless of code classification."

Specifically, LRA Section B.1.9 states that the BWR Stress Corrosion Cracking Program is an existing program that manages cracking of the reactor coolant pressure boundary using preventive measures, inspection, and flaw evaluation. In addition, LRA Section A.1.9 states that the program manages cracking of the reactor coolant pressure boundary using preventive measures, inspection, and flaw evaluation.

In addition, LRA Table 3.2.1, item 54, and LRA Table 3.3.1, item 110, indicate that the components in the engineered safety features and auxiliary systems, within the scope of the BWR Stress Corrosion Cracking Program, were reviewed as part of the Class 1 reactor coolant pressure boundary [as addressed in LRA Table 3.1.2-3].

Request.

a. Justify why LRA Sections B.1.9 and A.1.9 indicate that the BWR Stress Corrosion Cracking Program only manages aging of the RCPB rather than the recommended scope of the program, per the GALL Report, that includes relevant piping and piping welds regardless of code classification.

b. Clarify why LRA items 3.2.1-54 and 3.3.1-110 indicate that the components in the engineered safety features and auxiliary systems, subject to the program, were reviewed as part of the RCPB, rather than against the relevant piping and piping welds (regardless of code classification) that are included in the program scope.

c. If necessary, provide updates to LRA Sections B.1.9 and A.1.9 and items 3.2.1-54 and 3.3.1-110, consistent with the response.

RAI B.1.9-1a RESPONSE

The response to RAI B.1.9-1 stated that the BWR Stress Corrosion Cracking Program applies to BWR piping and piping welds regardless of code classification. That response also stated, at GGNS, all components included in the scope of the BWR Stress Corrosion Cracking Program are part of the reactor coolant pressure boundary and are subject to aging management review. To further clarify that response, there are no piping components or piping welds made of austenitic stainless steel or nickel alloy that are 4 inches or larger in nominal diameter and contain reactor coolant at a temperature above 93°C (200°F) during power operation in the non-Class 1 portions of the ESF and auxiliary systems (i.e., outside the reactor coolant pressure boundary).

a. LRA Sections A.1.9 and B.1.9 are revised to delete reference to reactor coolant pressure boundary and add reference to relevant piping and welds regardless of code classification.

b. LRA items 3.2.1-54 and 3.3.1-110 are revised to clarify that there are no components outside the reactor coolant pressure boundary subject to the BWR Stress Corrosion Cracking Program requirements.

c. See revised LRA Sections B.1.9 and A.1.9 and items 3.2.1-54 and 3.3.1-110.

Deletions are shown with strikethrough and additions are shown with underline.

LRA Section A.1.9 is revised to state:

A.1.9 <u>BWR Stress Corrosion Cracking Program</u>

The BWR Stress Corrosion Cracking Program manages cracking of ~~the reactor coolant pressure boundary~~ <u>relevant piping and piping welds regardless of code classification</u> using preventive measures, inspection, and flaw evaluation. Staff-approved BWRVIP documents and the GGNS response to NUREG-0313 Revision 2 and NRC Generic Letter 88-01 and its Supplement 1 are used to delineate the program.

LRA Section B.1.9 Program Description is revised to state:

B.1.9 BWR STRESS CORROSION CRACKING

<u>Program Description</u>

The BWR Stress Corrosion Cracking Program is an existing program that manages cracking of ~~the reactor coolant pressure boundary~~ <u>relevant piping and piping welds regardless of code classification</u> using preventive measures, inspection, and flaw evaluation. Staff-approved BWRVIP documents and the GGNS response to NUREG-0313 Revision 2 and NRC Generic Letter 88-01 and its Supplement 1 are used to delineate the program.

LRA items 3.2.1-54 and 3.3.1-110 are revised as shown below.

Table 3.2.1: Engineered Safety Features					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
3.2.1-54	Stainless steel piping, piping components, and piping elements exposed to treated water >60°c (>140°f)	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking," and Chapter XI.M2, "Water Chemistry"	No	This item was not used. Stainless steel components of the ESF systems subject to evaluation under the BWR Stress Corrosion Cracking Program were reviewed as part of the Class 1 reactor coolant pressure boundary (Table 3.1.2-3). There are no piping components or piping welds made of stainless steel that are 4 inches or larger in nominal diameter and contain reactor coolant at a temperature above 93°C (200°F) during power operation in the non-Class 1 portions of the ESF Systems.

In Item 3.2.1-54, "Class 1" was deleted in response to previous RAI B.1.9-1.

Table 3.3.1: Auxiliary Systems					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
3.3.1-110	Stainless steel piping, piping components, and piping elements exposed to treated water >60°c (>140°f)	Cracking due to stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking," and Chapter XI.M2, "Water Chemistry"	No	This item was not used. Stainless steel components of the auxiliary systems subject to evaluation under the BWR Stress Corrosion Cracking Program, were reviewed as part of the Class 1 reactor coolant pressure boundary. <u>There are no piping components or piping welds made of stainless steel that are 4 inches or larger in nominal diameter and contain reactor coolant at a temperature above 93°C (200 °F) during power operation in the non-Class 1 portions of the Auxiliary Systems.</u>

RAI B.1.9-2a

<u>Background</u>. By letter dated May 1, 2012, the applicant responded to RAI B.1.9-2, in part, to address the types of inspections of the stainless steel and nickel alloy thermal sleeve and sleeve extensions of reactor vessel nozzles (recirculation inlet, core spray inlet, and RHR/LPCI nozzles). In its response, the applicant indicated that the BWR Stress Corrosion Cracking Program [along with the Water Chemistry Program] is credited to manage cracking due to SCC and IGSCC in the thermal sleeves and thermal sleeve extensions of the reactor nozzles. The applicant's response also states that welds adjacent to specific components are inspected because welds are the susceptible areas.

In comparison, GALL Report item IV.B1.R-99 recommends the BWR Vessel Internals Program and Water Chemistry Program to manage cracking of the core spray nozzle thermal sleeves. In addition, Section 3.2.4, "Other Locations," of BWRVIP-18-A, "BWR Vessel and Internals Project BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines," indicates that there is currently no technique available for inspecting the core spray nozzle thermal sleeve welds. Inspection of thermal sleeve welds should be done when the capability exists.

<u>Issue</u>. The staff needs to further clarify whether the BWR Vessel Internals Program (including BWRVIP-18-A) is used to manage cracking of the core spray nozzle thermal sleeves as recommended in the GALL Report.

The staff also noted that BWRVIP-18-A indicates that there is currently no technique available for inspecting the thermal sleeve welds of the core spray nozzles and inspection of thermal sleeve welds should be done when the capability exists. It is not clear to the staff how the applicant's BWR Stress Corrosion Cracking Program inspects the thermal sleeves and thermal sleeve extensions to manage aging.

Request.

a. Provide justification for using the BWR Stress Corrosion Cracking Program to manage the aging of thermal sleeves and thermal sleeve extensions, given that they are typically located within the reactor vessel or piping. As part of the justification, describe how the BWR Stress Corrosion Cracking Program inspects these components (for example, using ultrasonic testing). In addition, describe the inspection results and operating experience in terms of occurrence of cracking in the thermal sleeves and sleeve extensions of the reactor nozzles.
b. Ensure that the LRA (including Table 3.1.2-1) is consistent with the applicant's response.

RAI B.1.9-2a RESPONSE

a. The thermal sleeves for the recirculation inlet, core spray inlet, and residual heat removal/ low pressure core injection (RHR/LPCI) nozzles are in part, formed by the internal leg of the Y shaped safe ends for those nozzles. The thermal sleeves are welded to the safe end and extend into the vessel. The welds connecting the safe ends to the vessel nozzles and external piping are reactor coolant pressure boundary welds. The BWR Stress Corrosion Cracking and Water Chemistry Control – BWR Programs manage cracking of these safe end pressure boundary welds. However, the safe ends for these nozzles and their pressure boundary welds are evaluated separately in LRA Table 3.1.2-1. The components represented by the lines in Table 3.1.2-1 for the thermal sleeves and extensions include the thermal sleeves starting from the weld to the safe end and extending inward. The thermal sleeves, starting from the weld to the safe end, are within the scope of the BWR Vessel Internals Program. GGNS operating experience is consistent with that reported in the related BWR Vessel Internals Program documents. Table 3.1.2-1 is revised to indicate that the BWR Vessel Internals and Water Chemistry Control – BWR programs manage cracking of the thermal sleeves and extensions.
b. Table 3.1.2-1 is revised as follows. Additions are marked with underline and deletions are marked with strikethrough.

Table 3.1.2-1: Reactor Vessel								
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**	**Item Number**	**Component**	**Aging Effect/ Mechanism**
Thermal sleeves ≥ 4" • Recirculation inlets (N2) • Core spray inlets (N5) • RHR/LPCI (N6)	Pressure boundary	Stainless steel	Treated water > 140°F (int)	Cracking	~~BWR Stress Corrosion Cracking~~ BWR Vessel Internals Water Chemistry Control – BWR	~~IV.A1.R-68~~ IV.B1.R-99 IV.B1.R-100	3.1.1-103~~97~~	~~A~~C
Thermal sleeve extensions ≥ 4" • Recirculation inlets (N2) • Core spray inlets (N5) • RHR/LPCI (N6)	Pressure boundary	Nickel alloy	Treated water (int)	Cracking	~~BWR Stress Corrosion Cracking~~ BWR Vessel Internals Water Chemistry Control – BWR	~~IV.A1.R-68~~ IV.B1.R-99 IV.B1.R-100	3.1.1-103~~97~~	~~A~~C

RAI B.1.9-3

Background. By letter dated May 1, 2012, the applicant responded to RAI B.1.9-2 to, in part, to address the types of inspections of the stainless steel and nickel alloy thermal sleeve and sleeve extensions of reactor vessel nozzles (recirculation inlet, core spray inlet, and RHR/LPCI nozzles).

As part of its response, the applicant identified a program exception. The applicant indicated that the use of a risk-informed inservice inspection application for Category A welds should be considered an exception since it is not expressly identified as an acceptable approach in NUREG-1801, Section XI.M7. The applicant also indicates that this exception is justified because the risk informed application was authorized by the NRC staff in a Safety Evaluation Report (SER), "Safety Evaluation by the Office Of Nuclear Reactor Regulation Request For Alternative GG-ISI-002 To Implement Risk-Informed Inservice Inspection (ISI) Program Based on The American Society Of Mechanical Engineers Code, ASME Code Case N-716," dated September 21, 2007.

The staff noted that the NRC letter dated September 21, 2007, states that the NRC staff authorizes the proposed alternative in accordance with 10 CFR 50.55a(a)(3)(i) for the remainder of the licensee's second 10-year ISI interval, which was extended by the NRC letter dated February 13, 2007, until fall 2008, and for its third 10-year ISI interval ending in 2017. The NRC letter also states that the NRC staff's approval of the licensee's RIS_B program does not constitute approval of Code Case N-716.

Issue. The staff noted that the NRC approval for the applicant's use of the ASME Code Case N-716 methodology, modified as described by the applicant's submittals, is granted for a certain portion of the second inservice inspection interval and the third interval only. If the applicant further pursues, the applicant would be required to reapply for the use of its risk-informed inservice inspection methodology for any 10-year interval beyond the third interval. Therefore, the staff needs to clarify what inspection scope and schedule the applicant would use in the BWR Stress Corrosion Cracking Program in the case the applicant could not get NRC approval for the use of the applicant's methodology.

Request. Clarify what inspection scope and schedule the BWR Stress Corrosion Cracking Program would use for Category A welds in the case the applicant could not get NRC approval for the use of the applicant's risk-informed inservice inspection methodology. Ensure that LRA Sections B.1.9 and A.1.9 are consistent with the response.

RAI B.1.9-3 RESPONSE

As stated in the response to RAI B.1.9-2, GGNS selects components and piping for examination based on the staff-approved inspection schedule and methods in BWRVIP-75-A. In the case that GGNS could not get NRC approval for the use of the risk-informed inservice inspection methodology, the inspection scope and schedule for the Category A welds in the BWR Stress Corrosion Cracking Program would follow BWRVIP-75-A which is consistent with NUREG-1801, Section XI.M7 BWR Stress Corrosion Cracking guidelines. LRA Sections A.1.9 and B.1.9 are revised as follows with additions in underline.

A.1.9 BWR Stress Corrosion Cracking Program

The BWR Stress Corrosion Cracking Program manages cracking of the reactor coolant pressure boundary using preventive measures, inspection, and flaw evaluation. Staff-approved BWRVIP documents and the GGNS response to NUREG-0313 Revision 2 and NRC Generic Letter 88-01 and its Supplement 1 are used to delineate the program. GGNS selects Category A welds for inspection in accordance with a risk-informed inservice inspection method. In the event a risk-informed inservice inspection method is not approved for use at GGNS, the inspection scope and schedule for Category A welds will be in accordance with BWRVIP-75-A.

B.1.9 BWR STRESS CORROSION CRACKING

Program Description

The BWR Stress Corrosion Cracking Program is an existing program that manages cracking of the reactor coolant pressure boundary using preventive measures, inspection, and flaw evaluation. Staff-approved BWRVIP documents and the GGNS response to NUREG-0313 Revision 2 and NRC Generic Letter 88-01 and its Supplement 1 are used to delineate the program. In the event a risk-informed inservice inspection method is not approved for use at GGNS, the inspection scope and schedule for Category A welds will be in accordance with BWRVIP-75-A.

RAI B.1.9-4

Background. Event Notification Report No. 47880, dated April 30, 2012, indicates that the applicant detected an unacceptable weld indication by ultrasonic testing in one of the residual heat removal (RHR) system to reactor pressure vessel nozzles (N06B-KB weld) during the current refueling outage. The dimension of the indication is approximately 0.9 inches in length and approximately 0.5 inches in depth. Nominal wall thickness of the weld is 1.3 inches.

The event notification also indicates that the weld defect has been evaluated by Entergy Engineering and determined to meet the criteria for reporting identified in NUREG-1022: Welding or material defects in the primary coolant system that cannot be found acceptable under ASME Section XI, IWB-3600, "Analytical Evaluation of Flaws," or ASME Section XI, Table IWB-3410-1, "Acceptable Standards."

Issue. The staff needs to confirm that this operating experience does not affect the effectiveness of the applicant's BWR Stress Corrosion Cracking Program.

Request.

a. Clarify whether the weld addressed in the event notification is included in the scope of the BWR Stress Corrosion Cracking Program. If this weld is included in the program scope, identify the material and category of the weld in accordance with Generic Letter 88-01 as referenced in the GALL Report (e.g., Alloy 82/182 weld with Category C prior to this event).

b. Evaluate this operating experience in terms of the effectiveness of the applicant's aging management as follows:
   1. Describe the results of the previous inspections performed as part of the BWR Stress Corrosion Cracking Program.

2. Describe the root or apparent cause analysis results and the corrective action to be taken. In addition, identify the new weld category assigned after this event in order to confirm whether it is consistent with Generic Letter 88-01 as referenced in GALL Report AMP XI.M7.
3. Justify why this operating experience does not affect the effectiveness of the applicant's program. As part of the response, identify any impact of this operating experience on the program elements of the applicant's BWR Stress Corrosion Cracking Program.

RAI B.1.9-4 RESPONSE

a. The N6B nozzle connects the residual heat removal C system to the reactor pressure vessel. The ISI weld N6B-KB is included in the scope of the BWR Stress Corrosion Cracking program. The 2012 examination was performed to comply with BWRVIP-75-A for Category C welds.

   The Category C N6B-KB RHR/LPCI inlet nozzle to safe end weld is constructed of the following materials. The nozzle material is carbon steel A-508 Class 2. The safe end material is Inconel SB-166 considered A600. The weld butter is Inconel 182 weld metal. The weld is Inconel 82/182 weld metal.

b1. Prior to 2012, no IGSCC type indication had been observed in the N6B nozzle. Previous exams were limited on the downstream side of the N6B-KB weld (nozzle side) due to the nozzle configuration, especially when performing a circumferential scan looking for axial flaws. The circumferential scan looking for axial flaws was only attempted once and was limited due to the weld crown and nozzle configuration. During 2012, this nozzle received extensive weld crown reduction and surface preparation which enabled the examination to detect the subject flaw.

   In 1990, induction heating stress improvement (IHSI) was performed on all dissimilar metal (DM) welds to perform stress relieving and prevent or mitigate nozzle cracks. The IHSI treatment on the DM welds was performed on the N6B nozzle. A post-ultrasonic test (UT) examination was completed and no indications were identified. However, as noted earlier, no circumferential scans were performed on the downstream side of the weld due to the configuration.

b2. The apparent cause of the weld indication is that the weld and weld butter were fabricated with material susceptible to IGSCC cracking. In 1990, actions were taken to mitigate this condition through the stress relieving process of induction heating stress improvement (IHSI). The cause for the failure to identify the indication prior to 2012 is the lack of improved ultrasonic examination techniques. After discovery of the weld indication in 2012, the weld was repaired by weld overlay. Prior to this event, the weld was Category C, as defined in BWRVIP-75-A. Future inspections of this weld will be conducted per BWRVIP-75-A for Category E welds.

   BWRVIP-61, "Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants", documents the results of an investigation into the most likely causes for post-IHSI IGSCC. Per the report, cracking reported after IHSI can be attributed to existing cracking that initially went undetected following application of IHSI. Flaws existed at the time of IHSI treatment, but were later detected with better examination methods and better trained personnel.

b3. Grand Gulf has 34 DM welds in the ISI program which require inspection under BWRVIP-75-A. Due to industry concerns with cracking in DM welds, the industry committed to an accelerated inspection program. Details of this program are provided in BWRVIP-222, "Accelerated Inspection Program for BWRVIP-75-A Category C Dissimilar Metal Welds Containing Alloy 182", July 2009. Twenty (20) DM Category C welds were inspected prior to 2012 to comply with the BWRVIP-222 requirements which became effective in 2002. The remaining fourteen (14) welds in the program were inspected during 2012. All BWRVIP-75-A Category C welds have been inspected. The N6B nozzle was the last nozzle requiring inspection.

The GGNS BWR Stress Corrosion Cracking Program is based upon the guidelines of BWRVIP-75-A. As stated in BWRVIP-75-A, since GL 88-01 was issued, the BWR industry has performed several thousand weld examinations on piping subject to the generic letter requirements. During this time, the industry has improved the water chemistry of reactor coolant thereby reducing initiation and growth of IGSCC. Stress improvement has also been employed as an IGSCC remedy. Examination procedures have been, and continue to be, improved. Examination personnel have received training on the latest techniques and have gained years of experience in the detection and sizing of IGSCC. Sufficient examinations have now been performed so that those welds that might have had shallow undetected cracking in 1988 have now been re-examined with qualified methods ensuring pipe integrity and a low likelihood of IGSCC.

The detection of this indication as described above demonstrates the effectiveness of the program in identifying IGSCC using today's improved inspection techniques and procedures. This operating experience has no impact on the program elements of the GGNS BWR Stress Corrosion Cracking Program. Continued implementation of the program provides reasonable assurance that the effects of aging will be managed so that components crediting this program can perform their intended function consistent with the current licensing basis during the period of extended operation.

RAI B.1.10-1a

Background and Issue. In RAI B.1.10-1, the staff requested that the applicant provide reference to the specific BWRVIP document credited for the BWR Vessel Attachment Welds Program. By letter dated May 1, 2012, the applicant responded to state that LRA Section A.1.10 was not changed to include the reference of BWRVIP-48-A. The applicant stated that the existing Section A.1.10 references "applicable industry standards and staff-approved BWRVIP documents," which provides a more comprehensive definition of applicant guidance to ensure program effectiveness than to list specific BWRVIP documents that may be revised or superseded in the future. This is contradictory to SRP-LR Table 3.0-1, "FSAR Supplement for Aging Management of Applicable Systems," for GALL Report AMP XI.M4, which specifically references BWRVIP-48-A.

10 CFR 54.21(d) requires that the UFSAR supplement contained a summary description of the program and activities for managing the effects of aging. Without an explicit reference to the appropriate document (i.e., BWRVIP-48-A) the summary description proposed by the applicant

is vague and does not allow the staff to make a finding of reasonable assurance regarding whether the proposed UFSAR supplement in LRA Section A.1.10 reflect an accurate summary description of the program and activities for managing the effects of aging.

Request. Revise LRA Section A.1.10 to indicate that the BWR Vessel Attachment Welds Program perform inspections and flaw evaluation in accordance with the guidelines in the BWRVIP-48-A report consistent with the SRP-LR's FSAR supplement. Alternatively, identify the section of the current UFSAR that references the BWRVIP-48-A report.

RAI B.1.10-1a RESPONSE

LRA Sections A.1.10 and B.1.10 are revised to include a reference to BWRVIP-48-A as shown below with additions underlined.

A.1.10 BWR Vessel ID Attachment Welds Program

The BWR Vessel ID Attachment Welds Program manages cracking in structural welds for BWR reactor vessel internal integral attachments using inspection and flaw evaluation in conformance with the guidelines of BWRVIP-48-A. ~~Applicable industry standards and staff-approved BWRVIP documents are used to delineate the program.~~

B.1.10 BWR VESSEL ID ATTACHMENT WELDS

Program Description

The BWR Vessel ID Attachment Welds Program is an existing program that manages cracking in structural welds for BWR reactor vessel internal integral attachments using inspection and flaw evaluation in conformance with the guidelines of BWRVIP-48-A. ~~Applicable industry standards and staff-approved BWRVIP documents are used to delineate the program.~~