Progress Energy

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Serial: HNP-05-093
10 CFR 50.55a

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

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/LICENSE NO. NPF-63

RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE RELIEF REQUEST FOR THE RISK-INFORMED ISI PROGRAM

Ladies and Gentlemen:

On June 21, 2005 and July 07, 2005, based on discussions with the NRC, the NRC requested additional information to facilitate the review of the proposed relief request (HNP-05-049 dated April 27, 2005) to use a Risk-Informed Inservice Inspection (RI-ISI) program for Class 1 and 2 piping welds at the Harris Nuclear Plant (HNP).

Attachment 1 provides the requested additional information from the first RAI (June 21, 2005).

Attachment 2 provides the requested additional information from the second RAI (July 07, 2005).

This document contains no new regulatory commitment.

Please refer any question regarding this submittal to Mr. Dave Corlett at (919) 362-3137.

Sincerely,

C.S. Kamilaris

C. S. Kamilaris
Manager, Support Services
Harris Nuclear Plant

CSK/jpy

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A047

Attachments:

1. Response to the First Request for Additional Information (RAI) Regarding the Relief Request for the Risk-Informed Inservice Inspection (ISI) Program

2. Response to the Second Request for Additional Information (RAI) Regarding the Relief Request for the Risk-Informed Inservice Inspection (ISI) Program

c:
Mr. R. A. Musser, NRC Senior Resident Inspector
Mr. C. P. Patel, NRC Project Manager
Dr. W. D. Travers, NRC Regional Administrator

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From the First Request for Additional Information (RAI) dated June 21, 2005:
Request: 1.

On page A2-4 of your submittal, you describe an industry probabilistic risk assessment peer review certification process that was performed in June 2002 on the Harris Nuclear Plant probabilistic safety assessment model. Per Regulatory Guide 1.178 dated September 2003, please list all Level A and B "Facts and Observations" from the review and how they have been addressed in the model. If some of the Level A and B "Facts and Observations" have not been resolved, please state why they are not expected to result in model changes that could significantly affect the overall results or conclusions of the RI-ISI consequence evaluation.

Response 1:

Table 1, List of All Harris Nuclear Plant's (HNP's) Level A and B Facts and Observations (F&Os) from the June 2002 Peer Review, provides a listing of all Level A and B F&Os and how they have been addressed in HNP's Probabilistic Safety Assessment (PSA) model of record (MOR2003).

Table 2, Harris Nuclear Plant's (HNP's) Open Facts and Observations (F&Os) from the June 2002 Peer Review, provides the open F&Os from the peer review and their proposed resolution.

All Level A and B F&O's from the June 2002 peer review have been reviewed for possible impact on the Risk Informed Inservice Inspection (RI-ISI) results and conclusions. All of the Level A and the majority of Level B F&Os were resolved and incorporated into the HNP PSA model of record or its associated documentation prior to use of the model in the RI-ISI consequence evaluation. None of the remaining open F&Os are expected to significantly impact the model, thus they are not expected to significantly impact the overall results and conclusions of the RI-ISI consequence evaluation. Therefore, the alternative proposed in HNP-05-049 dated April 27, 2005 would provide an acceptable level of quality and safety. The following summarizes the results of this review.

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From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

In the June 2002 peer review, two F&Os were categorized as Level A and 27 F&Os were categorized as Level B. The Level A F&Os were resolved and incorporated into the HNP PSA model of record that was subsequently used in the RI-ISI consequence evaluation. The majority of Level B F&Os (19 of 27) were resolved and incorporated into the HNP PSA model of record, or its associated documentation, that was subsequently used in the RI-ISI consequence evaluation. The following details the disposition of the Level B F&Os:

a. Eight F&Os were resolved by incorporating their associated changes into the HNP PSA model of record that was subsequently used in the RI-ISI consequence evaluation.

b. Eleven F&Os were resolved by adding additional information to the HNP PSA model of record supporting documents, but no changes were required to be made to the model that was subsequently used in the RI-ISI consequence evaluation.

c. The remaining eight Level B F&O responses remain open, and the specific observations from the peer review and their proposed resolution are detailed on Table 2.

The remaining eight Level B F&Os are planned to be resolved by the end of this year (2005). These F&Os cover the following areas: 1) Room heat-up, 2) Human Reliability Analysis (HRA) update, 3) Internal flooding, and 4) Inter-system LOCA (ISLOCA). These F&Os are not expected to result in model changes that could significantly affect the overall results or conclusions of the RI-ISI consequence evaluation. These F&Os are discussed in more detail below.

1) Room heat-up

   Results of the F&O relating to Room heat-up (F&O TH-01) identified that an additional operator action and plant procedural change (to open the switchgear room doors on loss of both trains of HVAC) were believed necessary to resolve the item. Plans are to incorporate this action into the PSA model, if required, during an update by the end of this year (2005). Modeling of this operator action is expected to have an insignificant impact on the PSA MOR2003 model results as applied to the RI-ISI consequence evaluation.

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From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

2) HRA update

The open F&Os relating to HRA (F&O's HR-01, HR-06, HR-09, HR-10, & HR-11) deal primarily with updating documentation to support the model HRA values. One F&O (F&O HR-10) recommends a review and update of dependent operator actions be performed and implemented in the PSA model. Since the HRA F&Os identified relate to improving documentation or update of dependencies, no significant changes to the PSA model are expected. A review and update of modeled operator actions is currently in progress with plans to incorporate changes into the PSA model during an update by the end of this year (2005). A sensitivity study shows that increasing all current HRA values by 50% (excluding internal flooding actions) will only increase Core Damage Frequency (CDF) by approximately 12% and Large Early Release Frequency (LERF) by approximately 13%. It is unlikely that resolution of the HRA F&Os would cause this magnitude of change to the HRA values. Based on the sensitivity performed, it is not expected that changes in modeled operator actions will have a significant impact on the PSA MOR2003 model results as applied to the RI-ISI consequence evaluation.

3) Internal flooding

The open F&O related to Internal Flooding (F&O DE-10) relates to providing additional documentation to support analysis results. A re-analysis for the internal flooding contribution to core damage frequency and large-early-release frequency shows the results to be approximately the same as previous analysis results. Therefore, it is not expected that internal flooding operator actions will have a significant impact on the PSA MOR2003 model results as applied to the RI-ISI consequence evaluation.

4) Inter-System LOCA (ISLOCA)

The remaining outstanding F&O deals with ISLOCA (F&O ST-01) stating that the analysis does not consider the probability of failure of the RHR Heat Exchanger Channel shell and that the ISLOCA frequency could be increased by an order of magnitude. Sensitivity studies show that increasing the PSA MOR2003 ISLOCA frequencies by an order of magnitude yields a small increase in CDF of 0.12 percent and in LERF of 0.76 percent which should have an insignificant impact on the PSA MOR2003 model results as applied to the RI-ISI consequence evaluation.

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From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

In summary, since resolution of the open F&Os is not expected to significantly impact the HNP PSA model of record (MOR2003) results, then none of the open F&Os are expected to significantly impact the overall results and conclusions of the RI-ISI consequence evaluation. Therefore, the alternative proposed in HNP-05-049 dated April 27, 2005 would provide an acceptable level of quality and safety.

From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

TABLE 1 **List of All Harris Nuclear Plant's (HNP's) Level A and B** **Facts and Observations (F&Os) from the June 2002 Peer Review**		
F&O No.	F&O Level	How the F&O Was Addressed
AS-07	A	Incorporated into the PSA model of record (MOR2003).
QU-01	A	Incorporated into the PSA model of record (MOR2003).
IE-07	B	Incorporated into the PSA model of record (MOR2003).
IE-08	B	Incorporated into the PSA model of record (MOR2003).
AS-03	B	Incorporated into the PSA model of record (MOR2003).
DA-01	B	Incorporated into the PSA model of record (MOR2003).
DA-04	B	Incorporated into the PSA model of record (MOR2003).
DA-06	B	Incorporated into the PSA model of record (MOR2003).
HR-04	B	Incorporated into the PSA model of record (MOR2003).
QU-02	B	Incorporated into the PSA model of record (MOR2003).
IE-05	B	No model changes were required, but supporting documentation was updated.
IE-09	B	No model changes were required, but supporting documentation was updated.
AS-01	B	No model changes were required, but supporting documentation was updated.
AS-06	B	No model changes were required, but supporting documentation was updated
SY-08	B	No model changes were required, but supporting documentation was updated
SY-11	B	No model changes were required, but supporting documentation was updated
SY-12	B	No model changes were required, but supporting documentation was updated
DA-02	B	No model changes were required, but supporting documentation was updated
HR-07	B	No model changes were required, but supporting documentation was updated
DE-11	B	No model changes were required, but supporting documentation was updated
QU-07	B	No model changes were required, but supporting documentation was updated
TH-01	B	Open F&O (Subject: Room Heat-up).
HR-01	B	Open F&O (Subject: Human Reliability Analysis)
HR-06	B	Open F&O (Subject: Human Reliability Analysis)
HR-09	B	Open F&O (Subject: Human Reliability Analysis)
HR-10	B	Open F&O (Subject: Human Reliability Analysis)
HR-11	B	Open F&O (Subject: Human Reliability Analysis)
DE-10	B	Open F&O (Subject: Internal Flooding).
ST-01	B	Open F&O (Subject: Inter-System Loss of Coolant Accident).

From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

TABLE 2
Harris Nuclear Plant's (HNP's) Open Facts and Observations (F&Os) from the June 2002 Peer Review

F&O No.	F&O	Proposed Resolution
TH-01	Room Heat-up: "Shearon Harris has a room heatup calculation that addresses room heatup for the Charging/Safety Injection Pump Rooms, Switchgear Rooms, ESW Pump Rooms and RAB 236' Elevation. Additionally the calculation states that the requirement for EDG Room Cooling is required. Additional justification/reference should be given for the screening out of additional plant areas. In the calculation of the room heatup for RAB 236' elevation, the equipment assumed to be in operation is limited to 2 CCW pumps and heat exchangers, one service water booster pump and one motor driven auxiliary feedwater pump. Additionally in the area are the second motor driven auxiliary feedwater pump and the turbine driven auxiliary feedwater pump. For certain accident sequences, specifically ATWS, all auxiliary feedwater pumps would be in operation and therefore the room heatup calculation would not be bounding. Further, for a SBO, the turbine driven feed pump, with its high heat loads, would be in operation and could fail prematurely due to high room temperatures."	A room heat-up analysis report, NAI-1169-001, has been completed for HNP. Justification for screening out plant areas is presented. Detailed screening of the CCW Pump Room (analysis includes 2 motor-driven auxiliary feedwater pumps, one turbine-driven auxiliary feedwater pump, 2 component cooling water pumps, one ESW booster pump, 2 CCW heat exchangers, piping, cable trays, and lighting) resulted in a maximum room temperature of 142.4 degrees Fahrenheit. This analysis bounds the ATWS and SBO scenarios, since only the turbine-driven auxiliary feedwater pump is in operation for SBO. The reference room heat-up analysis also identified the two switchgear rooms as needing operator action (to open doors) to maintain room temperatures at acceptable levels for equipment operation. Plans are to incorporate this action into the PSA model, if required, during an update in late 2005. Modeling of this operator action should have an insignificant impact on the MOR2003 model results and core damage frequency.

From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

TABLE 2
Harris Nuclear Plant's (HNP's) Open Facts and Observations (F&Os) from the June 2002 Peer Review

F&O No.	F&O	Proposed Resolution
HR-01	Human Reliability Analysis (HRA): "The source and basis for time windows for proceduralized operator actions should be clearly indicated (T&H analysis, generic reports, expert opinion, etc.). For this purpose, a summary table may be used. Current documentation should be supplemented to assure that time windows are systematically and consistently defined and used. The time windows are usually stated at the end of the operator action descriptions, giving the impression that they may not have been used in the analysis. It is imperative that time windows are sufficient to allow recovery credit and local action feasibility. RCP seal LOCA initiation at the time of 90 minutes is used as a basis for operator action time windows in various cases (see OPER-20, OPER-45). This time window, as discussed in F&O AS-03, should be reviewed and revised."	A review and update of all modeled operator actions is currently in progress. Plans are to incorporate any changes into the PSA model during an update in late 2005. It is not expected that changes in modeled operator actions will have a significant impact on the MOR2003 model results and core damage frequency.
HR-06	Human Reliability Analysis (HRA): "An error factor of 3 is applied to many HRA inputs throughout the analysis. This influences the mean values calculated from median parameter values. This relatively low range factor will lead to generally lower mean value estimates than would be calculated based on higher range factors. At a minimum, recognition of this impact should be included in the documentation. What about actions for which there is little basis for the 3, e.g., high stress, time critical actions? The state of uncertainty associated with some actions could warrant a significantly higher range factor."	A review and update of all modeled operator actions is currently in progress. Plans are to incorporate any changes into the PSA model during an update in late 2005. It is not expected that changes in modeled operator actions will have a significant impact on the MOR2003 model results and core damage frequency.

From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

TABLE 2
Harris Nuclear Plant's (HNP's) Open Facts and Observations (F&Os)
from the June 2002 Peer Review

F&O No.	F&O	Proposed Resolution
HR-09	Human Reliability Analysis (HRA): "The time available to complete post initiator human actions is discussed in some level of detail for each HI in Attachment C of Appendix F of the PSA. But the actual time available, and a link to a specific supporting MAAP analysis is difficult to trace. For example, OPER-3 states that the operators have a 45 minute window from the loss of feedwater until core damage is inevitable. The success criteria in Appendix D quotes a time window of 70 minutes after trip. For this action (bleed and feed) the time window for operator action is also impacted by the time spent by operators attempting to restore feedwater, prior to implementing bleed and feed. This impact on the time window is not clearly discussed. Given a time frame is established, there is no reference to the specific supporting MAAP analysis. The MAAP analyses that support the HRA timing and accident sequence success criteria have been updated recently, based on power uprate and SG replacement, and HNP PSA staff reviewed the new MAAP results with respect to potential impacts on the PSA, but this review is not documented as part of the model of record."	A review and update of all modeled operator actions is currently in progress. Plans are to incorporate any changes into the PSA model during an update in late 2005. It is not expected that changes in modeled operator actions will have a significant impact on the MOR2003 model results and core damage frequency.

From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

TABLE 2
Harris Nuclear Plant's (HNP's) Open Facts and Observations (F&Os) from the June 2002 Peer Review

F&O No.	F&O	Proposed Resolution
HR-10	Human Reliability Analysis (HRA): "The HNP PSA documents a procedure for identifying potentially dependent HEPs that occur in the same cutset in the quantified results. This procedure is documented in Section E.1, and the implementation of the procedure is also documented. The procedure is good, but has not been implemented for the current model of record. Even without an update of the HRA analysis, the search for dependent HEPs in the cutsets should be repeated as part of the model quantification procedure."	A review and update of all modeled operator actions is currently in progress. Plans are to incorporate any changes into the PSA model during an update in late 2005. It is not expected that changes in modeled operator actions will have a significant impact on the MOR2003 model results and core damage frequency.
HR-11	Human Reliability Analysis (HRA): "The[re] is no evidence that the results of the HRA, or even the risk significant HEPs, were reviewed by plant operators."	A review and update of all modeled operator actions is currently in progress. Plans are to incorporate any changes into the PSA model during an update in late 2005. It is not expected that changes in modeled operator actions will have a significant impact on the MOR2003 model results and core damage frequency.
DE-10	Internal Flooding: "The internal flooding (IF) analysis utilizes 3 HEPs which are described in the HRA notebook as screening values, and references the IF notebook for their development. While the IF notebook describes the mitigation events for which the HEPs values are provided, neither the specific bases or sources for the values nor PSFs utilized are provided. Screening values are not appropriated for these important contributors to risk."	A detailed HRA evaluation has been completed and included in the internal flooding analysis. Design Calculation HNP-F/PSA-0057, Revision 2 incorporates these changes. Plans are to incorporate these changes into the PSA model during an update in late 2005. Changes to the flooding HRA are not expected to have a significant impact on the MOR2003 model results and core damage frequency.

From the First Request for Additional Information (RAI) dated June 21, 2005:
Response 1 (continued):

TABLE 2 Harris Nuclear Plant's (HNP's) Open Facts and Observations (F&Os) from the June 2002 Peer Review		
F&O No.	F&O	Proposed Resolution
ST-01	Inter-System Loss of Coolant Accident (ISLOCA): "The ISLOCA analysis does not consider the probability of failure of the RHR Heat Exchanger channel shell or head (i.e., that portion of the heat exchanger that is exposed to reactor coolant before and after if flows through the tubes). At least one report (NUREG/CR-5744) indicates the possibility of this being more probable than the piping."	The evaluation for this item has not been completed. However, Sensitivity studies show that increasing the MOR2003 ISLOCA frequencies by an order of magnitude yields an increase in Core Damage Frequency of only 0.15 percent and an increase in Large Early Release Frequency of only 0.76% which should have an insignificant impact on the PSA MOR2003 model results.

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RESPONSE TO THE SECOND REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE RELIEF REQUEST FOR THE RISK-INFORMED INSERVICE INSPECTION (ISI) PROGRAM

From the Second Request for Additional Information (RAI) dated July 07, 2005:
Request: 1.

In Articles IWB-1000 and IWC-1000 of ASME Code Section XI, certain components are exempted from the volumetric and surface examination requirements of IWB-2500 and IWC-2500, respectively. Confirm that the exempted components are evaluated in the licensee's RI-ISI program. Provide justification if those exempted components are not evaluated in the RI-ISI program as some of these exempted components may have high or medium risk classifications which would require some examinations to be performed.

Response 1:

The Risk-Informed Inservice Inspection (RI-ISI) evaluation scope for Harris Nuclear Plant (HNP) was defined consistent with the existing plant ISI Program examination boundaries for Class 1 and Class 2 piping. The Class 1 and Class 2 piping that is exempted from examination in the existing ISI Program for HNP according to IWB-1220 and IWC-1220 is also considered exempt from RI-ISI Program requirements.

The use of Code exemptions for defining the RI-ISI evaluation scope has been consistently applied in all follow-on plant applications that have implemented the EPRI RI-ISI methodology, and these plant applications have been approved by the NRC.

In addition, Appendix R entitled "Risk-Informed Inspection Requirements for Piping" is included in the 2005 Addenda of the ASME Section XI Code. An excerpt is provided below from Supplement 2 (Section 2.0 - Boundary Identification) of Appendix R, which permits the use of Code exemptions to define the RI-ISI scope. Supplement 2 provides the Risk Informed Selection Process for Method B (EPRI).

> "Piping, or portions thereof, included for evaluation shall be based on the deterministic program Class 1, 2, or 3, examination boundaries, if applicable, determined in accordance with the requirements of IWA-1320, and limited by exemptions of IWB-1220, IWC-1220, and IWD-1220. When Examination Category C-F-1 or C-F-2 piping is included, the piping exempt from NDE under the requirements of Table IWC-2500-1 due to nominal wall thickness limitations shall be evaluated."

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From the Second Request for Additional Information (RAI) dated July 07, 2005:
Request: 2.

In Section 3, Risk-Informed Process, the licensee discussed a deviation to the EPRI RI-ISI methodology in the failure potential assessment for thermal stratification, cycling and striping (TASCS). On page A2-8, the licensee stated that the final materials reliability program (MRP) guidance on the subject of TASCS will be incorporated into the HNP RI-ISI application if warranted. Since the final MRP guidance on TASCS has not been issued, confirm that only the portions of the final MRP guidance that are reviewed and approved by NRC will be incorporated into the licensee's RI-ISI program.

Response 2:

HNP will review the final approved MRP-146, Materials Reliability Program Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines, and the NRC limitations on its use. HNP will either incorporate the portions of the MRP guidance that are reviewed and approved by the NRC or provide justification for relief in accordance with 10 CFR 50.55a(g)(5)(iii).

Request 3:

In Table 3.6.1, Risk Impact Analysis Results (Attachment 2), Risk Categories 5a, 6a and 7a were assigned for some piping system segments. Risk Categories 1 through 7 are discussed in the EPRI methodology for RI-ISI program. Provide additional information regarding the Risk Categories 5a, 6a and 7a as referenced in Table 3.6.1, since they were not explicitly discussed in the licensee's RI-ISI program.

Response 3:

Letter designations are used in the HNP RI-ISI Program for Risk Categories 5, 6 and 7 as indicated in the Risk Matrix below. These letter designations are used strictly for data management purposes and have no bearing on the RI-ISI Program for HNP. The use of these letter designations permits an immediate identification of the specific Risk Matrix "bin" each weld location is assigned to in those cases where different combinations of consequence and failure potential rank results can produce the same Risk Category (i.e., 5, 6 and 7).

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From the Second Request for Additional Information (RAI) dated July 07, 2005:
Response 3 (continued):

HNP Risk Matrix		**Conditional Core Melt Potential**			
		None	**Low**	**Medium**	**High**
Pipe Rupture Potential	**High**	Risk Category 7b	Risk Category 5b	Risk Category 3	Risk Category 1
	Medium	Risk Category 7c	Risk Category 6b	Risk Category 5a	Risk Category 2
	Low	Risk Category 7d	Risk Category 7a	Risk Category 6a	Risk Category 4

Request 4:

In Table 3.3, Failure Potential Assessment Summary (Attachment 2), the staff notes that PWSCC is identified only in the reactor coolant system. Since Inconel weld is susceptible to PWSCC, confirm that there is no Inconel weld in any other piping systems that are in scope of the RI-ISI program.

Response 4:

HNP is a Westinghouse designed pressurized water reactor. The reactor coolant system is the only piping system in the scope of the RI-ISI Program that contains inconel material (Alloy 182, 82, 52). Inconel material is present at the Hot and Cold Leg nozzle connections (Alloy 182, 82) to the Reactor Vessel and the Pressurizer Surge, Spray, Safety and Relief Valve nozzle connections (Alloy 182, 82). The Hot and Crossover Leg nozzle connections to the Steam Generators are Alloy 52.

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From the Second Request for Additional Information (RAI) dated July 07, 2005:
Request 5:

In accordance with Table 3.5 (Attachment 2), the staff notes that the majority of the elements selected for inspection are in Risk Category 4. Since Category 4 does not identify any degradation mechanism, provide some discussion including the bases regarding what examination volume and method will be used for the inspection of the selected elements.

Response 5:

All Risk Category 4 element locations selected for inspection at HNP will be subject to volumetric examination. The examination volume will be expanded beyond that required by the ASME Section XI Code to include the counterbore transition region.

The performance of a volumetric examination on an expanded examination volume for Risk Category 4 inspection locations is consistent with the requirements of Table R-2500-1 in Appendix R of the ASME Section XI Code that is discussed above in the response to Request 1. Item No. R1.20 in Table R-2500-1 specifies the above examination requirements for elements not subject to a degradation mechanism (i.e., Risk Category 4).

Request 6:

In Section 2.2, Augmented Programs (page A2-5), the licensee stated that a plant augmented inspection program is currently implemented in response to SER Question 250.1 that requires volumetric or surface examination of thin-walled charging, containment spray, residual heat removal and safety injection system piping welds. The licensee also concluded that the RI-ISI program effectively subsumes this plant augmented inspection program. Please provide detailed information and discussion regarding what is SER Question 250.1 and how the assessment was performed to support the stated conclusion. Identify any precedents where NRC has approved the subsumption of this plant augmented inspection program.

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From the Second Request for Additional Information (RAI) dated July 07, 2005:
Response 6:

A plant augmented inspection program is currently implemented at HNP in response to SER Question 250.1 that requires volumetric or surface examination of thin-walled charging, containment spray, residual heat removal and safety injection system piping welds. The purpose of this plant augmented inspection program for the subject system piping is two-fold as discussed in (a) and (b) below.

(a) The piping of concern addressed by IE Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants", is included in the scope of the plant augmented inspection program implemented in response to SER Question 250.1. The intergranular stress corrosion cracking (IGSCC) concern addressed by IE Bulletin 79-17 was explicitly considered in the application of the EPRI RI-ISI process and is subsumed by the RI-ISI Program per Section 6.5 of EPRI TR-112657.

(b) This plant augmented inspection program also addresses thin-walled (< 3/8 inch nominal wall thickness) piping segments in the subject systems. Although these piping segments are not exempt per IWC-1220, they are not subject to nondestructive examination per the requirements of Table IWC-2500-1 due to nominal wall thickness. The thin-walled piping segments in the subject systems were assessed as part of the RI-ISI application scope for HNP. This is consistent with Appendix R of the ASME Section XI Code that is discussed above in the response to Question (1). An excerpt is provided below from Supplement 2 (Section 2.0 – Boundary Identification) of Appendix R that addresses the issue of thin-walled Class 2 piping.

> "When Examination category C-F-1 or C-F-2 piping is included, the piping exempt from NDE under the requirements of Table IWC-2500-1 due to nominal wall thickness limitations shall be evaluated."

As discussed in (a) and (b) above, the subject system piping addressed by the plant augmented inspection program in response to SER Question 250.1 has been assessed in the RI-ISI application scope for HNP. The IGSCC concern of IE Bulletin 79-17 was explicitly considered and all thin-walled piping segments were evaluated. All follow-on plant applications (including the pilot plant, ANO-2) with an IE Bulletin 79-17 augmented program have had the program subsumed in the manner described above. All follow-on plant applications with thin-walled piping segments have had these segments assessed as part of the RI-ISI application in the manner described above.