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

Joseph M. Farley Nuclear Plant
Risk-Informed Inservice Inspection Program Submittal
Response to a Request for Additional Information

Ladies and Gentlemen:

In a letter dated November 03, 2003, titled "Joseph M. Farley Nuclear Plant, Units 1 and 2 - Request for Additional Information Re: Risk Informed Inservice Inspection Program (TAC Nos. MC0178 and MC0179)," the NRC requested Southern Nuclear Operating Company (SNC) to provide additional information related to the SNC request on July 17, 2003 to use a Risk-Informed Inservice Inspection (RI-ISI) Program for certain ASME Code piping.

SNC's response to the request for additional information is enclosed.

As stated in SNC's FNP RI-ISI Program submittal letter NL-03-1259 dated July 17, 2003, SNC requests NRC approval of the FNP RI-ISI Program by December 31, 2003, in order to support implementation of the Program during the FNP Unit 2 2R16 Maintenance/Refueling Outage currently scheduled to begin in the spring of 2004.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

J. B. Beasley, Jr.

JBB/JLS/sdl

Enclosure: Response to Request for Additional Information

A047

cc: Southern Nuclear Operating Company
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Mr. D. E. Grissette, General Manager – Plant Farley
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U. S. Nuclear Regulatory Commission
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Mr. F. Rinaldi, NRR Project Manager – Farley
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Joseph M. Farley Nuclear Plant
Risk-Informed Inservice Inspection Program Submittal
Response to a Request for Additional Information

Enclosure

Response to Request for Additional Information

Joseph M. Farley Nuclear Plant
Risk-Informed Inservice Inspection Program Submittal
Response to a Request for Additional Information

Enclosure

Response to Request for Additional Information

By letter dated November 3, 2003, the NRC requested additional information for the Farley Risk-Informed ISI (RI-ISI) submittal dated July 17, 2003. Questions 1 through 6 pertain to failure probability estimates, risk ranking, and weld selection. Questions 7 through 9 pertain to the quality of the PRA model used for the RI-ISI process at Farley and more specifically pertain to the observations/comments from the peer review of the PRA model.

While a response to each RAI is given below, it is important to understand that for the PRA related questions, as a whole, the overall conclusion of the Westinghouse Owners Group peer review of the Farley PRA model was that it contained the necessary attributes required for risk ranking of systems, structures, and components. This peer review was conducted using a process consistent with Nuclear Energy Institute guidance published in NEI-00-02, "Industry Peer Review Process." During the course of the peer review, enhancements were identified in observations/comments such as those cited in questions 7 through 9, which would only require resolution to support use of the Farley PRA model for higher level risk-informed regulatory applications. This application is not considered a higher level risk-informed regulatory application. Therefore, even though a resolution has not been reached for all peer review observations/comments, the PRA model attributes important to the Westinghouse RI-ISI application are present in the Farley Revision 5 PRA model.

Question 1

SNC has reported that Resistance Temperature Detectors (RTDs) were installed on the top and bottom of unisolable reactor coolant system (RCS) branch piping identified in SNC's response to NRC Bulletin 88-08 for piping believed to be susceptible to thermal stratification and cycling. If the top-to-bottom temperature differential exceeds a predetermined value, the licensee determines the cause and the potential damage to the piping. For these monitored lines, SNC did not consider the potential for thermal stratification and cycling in the structural reliability and risk assessment (SRRA) failure probability assessments. Therefore, the potential for thermal stratification and cycling were not reflected in the calculated risk ranking values. The Interim Thermal Fatigue Management Guideline (MRP [Materials Reliability Program] -24) was used to screen those lines that were not monitored for potential thermal fatigue cracking.

The NRC staff agrees that implementation of continuous temperature monitoring can help reduce the probability of pipe failure caused by thermal stratification and cycling. However, it will not eliminate the potential occurrence of this degradation mechanism nor can it be expected to eliminate any degradation to the piping if the loading occurs. The extent of damage to the pipe will depend on a number of factors including: the cause of stratification condition, the cyclic nature of the mechanism (e.g., high or low cycle behavior caused by turbulent penetration effects, convection flow, etc.), the time before the corrective actions can be implemented to eliminate the stratification load, and the ability to characterize the extent of the cracking that might result during exposure to these loading conditions. In addition, synergistic effects resulting from thermal stratification degradation can affect failure probabilities for other cycle fatigue loadings. In light of these concerns, the NRC staff believes that the pipe segment failure probability assessments for the monitored piping and resulting segment risk ranking for these lines should include a contribution for thermal stratification and cycling. Please, discuss whether the SRRA models for these lines

need to be revised to include consideration for cyclic thermal stratification. If so, report the revised failure probability estimates and identify any changes to the risk ranking of these segments.

Response

Piping monitored at Farley for thermal stratification that is considered susceptible to thermal stratification and cycling induced cracking includes the unisolable (from the reactor coolant system) portions of the hot and cold leg injection lines and the Residual Heat Removal suction lines. The SRRA values for these segments do not need to be revised to account for thermal stratification and cycling because these segments are already designated as Highly Safety Significant (HSS) and are already scheduled for examination. The incremental change that would result from the contribution for thermal stratification and cycling would have no effect on the HSS risk ranking of these segments and is thus not warranted.

While cyclic thermal fatigue was not modeled, when nondestructive examinations are performed on each of these segments, the examination includes areas considered potentially susceptible to cyclic thermal stratification cracking.

Question 2

SNC has stated that for FNP, each unit has a total of 18 dissimilar metal welds. All the dissimilar metal welds are located in the RCS piping and are in contact with primary coolant. Six reactor pressure vessel (RPV) nozzle safe-end welds and six pressurizer nozzle safe-end welds contain Inconel 82 weld material. The remaining six dissimilar metal welds consist of Inconel 52 buttered hot and cold leg nozzles located on the steam generators (SG). Because of primary water stress corrosion cracking (PWSCC) issues associated with Inconel weld material in contact with reactor coolant, SNC has selected all Inconel 82 welds and three cold leg SG nozzle Inconel 52 welds for examination. SNC has stated that since Inconel 52 weld material is generally considered to be less susceptible to PWSCC, the remaining three SG Inconel 52 welds located in the same hot leg as the three Inconel 82 RPV outlet nozzle welds were not selected for examination.

Limited laboratory data suggests that Alloy 52 weld material offers improved resistance to PWSCC over Alloy 82 material. However, current understanding is based on a very limited amount of data on laboratory-prepared specimens. Also, very little service experience has been accumulated for these weld materials in thick section pressurized water reactor (PWR) reactor coolant piping. Recent investigations have found that many weldability issues associated with Alloy 52/152 thick welds are just beginning to be recognized. Significant amounts of ductility dip cracking, lack of fusion and porosity have been observed. Weldability issues like these have resulted in significant numbers of repairs and higher localized residual stresses at the inside surface of the weld. In a recent application, the NRC inspection team concluded that the PWSCC phenomenon for Alloy 52/152 welding material is not fully understood and further studies developing quantitative data should be performed before the new Alloy 52/152 weld can be considered immune to PWSCC. In light of the above discussion and in keeping with the fundamental defense-in-depth principals, the NRC staff believes that PWSCC should be treated as an "active" degradation mechanism for all 18 RCS dissimilar metal welds in each Farley unit and a "high failure importance" should be assigned to each of these welds. This is consistent with the definition for high failure importance in Westinghouse Report WCAP-14572, Revision 1, Section 3.7.1, as interpreted in Section 3.4.1 of the NRC Safety Evaluation Report, dated December 15, 1998. Provide further discussion on this issue as it relates to Category B-F welds at FNP, and show justification for not inspecting all dissimilar metal welds in high safety significant segments.

Response

SNC agrees to include the three additional Inconel 52 welds into the examination scope; therefore, all 18 dissimilar metal welds (located on 15 segments) will be in the ten-year, RI-ISI examination scope. A re-evaluation of any new quantitative data concerning the resistance of heavy wall Inconel 52 welds to PWSCC will be performed to determine if the three Inconel 52 welds should continue to be in the examination scope during the next 10-year interval update.

As discussed during the October 6, 2003 conference call between SNC and the NRC staff, SNC does not believe that treating each of the dissimilar metal welds as having an "active" degradation mechanism" and having "high failure importance" would benefit the objective of the RI-ISI process (i.e., elimination of dose and costs associated by reducing examinations while maintaining or enhancing safety). Treatment of these segments as requested by the NRC staff would require the examination of each dissimilar metal weld (as agreed to by SNC); however, the process would also require the examination of one additional non-Inconel, stainless steel weld in each of the 15 segments per each Farley unit. One of these segments (the pressurizer surge line) has a stainless steel weld that is required to be examined (in addition to the Inconel weld) because of postulated thermal fatigue. However, the examination of the remaining 14 RCS stainless steel welds, which do not have an identified active failure mechanism, would result in an increase in the associated dose (estimated to be approximately one man-Rem) and examination costs, without a corresponding benefit (little or no change to the delta risk calculations would occur).

Therefore, SNC's position is that we will include all 18 dissimilar metal welds in the ten-year, risk-informed ISI examination scope; however, it is also our position that we should not treat each of the 15 segments (containing the dissimilar metal welds) as having an active degradation mechanism and having high failure importance.

Question 3

SNC has committed to perform the examinations listed in Table 4.1-1 of the WCAP-14572, A-version (WCAP), with the exception of the examinations required for Primary Water Stress Corrosion Cracking (PWSCC). The WCAP lists a visual VT [Visual Testing] -2, performed during system or component pressure tests to detect PWSCC. SNC has noted that VT-2 tests are not volumetric, and as such, will implement VT-2 per ASME Section XI, Table IWB-2500-1. Also, SNC has committed to performing volumetric or "other appropriate examinations" each interval to detect PWSCC originating from the ID of susceptible piping.

The NRC staff contends, and SNC recognizes, that the visual VT-2 specified in Table 4.1-1 of the WCAP is not appropriate for detecting PWSCC prior to failure of the component having occurred. Given recent industry experience, it is expected that internally-initiated PWSCC will occur at the Inconel-bearing dissimilar metal welds (ASME Category B-F) exposed to primary coolant. Therefore, the guidelines in Table 4.1-1 of the WCAP are not acceptable for piping elements susceptible to PWSCC. SNC should confirm that all Category B-F welds susceptible to PWSCC will be volumetrically examined each interval as part of the RI-ISI program. In addition, the licensee should describe what is intended by "other appropriate examinations" that may be applied to these welds.

Response

All 18 Category B-F welds will be volumetrically examined each 10-year interval as part of the RI-ISI program. The use of the phrase "other appropriate examinations" referred to alternate examinations that may be accepted in the future by the NRC as an appropriate alternative (e.g., VT-1 of the weld from the inside surface).

Question 4

Table 3.4.1 indicates that failure probabilities (cumulative for 40 years) as high as 1.04E-01 (small leak) and 4.23E-02 (disabling leak) for the main steam system, and as high as 8.56E-02 (small leak) and 5.26E-02 (disabling leak) for the chemical and volume control system. Failure probabilities at such high levels would suggest that observable damage (small leaks, cracking, or wall thinning) has been observed at Farley or other plants with similar designs and operating conditions.

a) Describe the degradation mechanisms and locations in the main steam (MS) and chemical volume control system (CVCS) that correspond to these values of failure probabilities.

b) Describe applicable operating experience that would support the high values of failure probabilities listed in Table 3.4-1 of the WCAP for the MS and CVCS.

c) To what extent are the inspections for the MS and CVCS as listed in Table 5-1a and 5-1b of the WCAP, directed to the locations associated with the high values of failure probabilities listed in Table 3.4-1 of the WCAP?

Response

Main Steam System:

a) The high SRRA values are for eleven 1-inch socket-welded segments on Unit 2. Three of these segments are located inside containment and the remainder of the segments are located outside containment. The dominant degradation mechanism for these segments is high vibration.
b) Cracking of small main steam socket welded segments has occurred on Unit 2 due to high vibration; however, due to the low consequences resulting from a crack in these small lines, they are considered Low Safety Significant (LSS).
c) These segments are not associated with the "ASME Section XI" examinations listed in Tables 5-1a and 5-1b because they are exempt from the existing ASME Code requirements. Additionally, these segments are not associated with the "RI-ISI" examinations listed in Tables 5-1a and 5-1b because the table is for HSS segments and these are LSS segments.

Chemical Volume and Control System:

a) The high SRRA values are for 2-inch socket-welded LSS segments located in the vicinity of the letdown flow orifices. The dominant degradation mechanism for these segments is postulated vibration caused by the large pressure drop at the flow orifices.

b) There has not been a cracking issue at Farley in these segments; however, cracking has occurred in the nuclear industry. For additional information, see NRC Information Notice 98-45, "Cavitation Erosion of Letdown Line Orifices Resulting in Fatigue Cracking of Pipe Welds."
c) These segments are not associated with the "ASME Section XI" examinations listed in Tables 5-1a and 5-1b because they are exempt from the existing ASME Code requirements. Additionally, these segments are not associated with the "RI-ISI" examinations listed in Tables 5-1a and 5-1b because the table is for HSS segments and these are LSS segments.

Question 5

The licensee argues that, in several instances, the RI-ISI program will require examinations that are not currently required by the ASME Section XI program. Examples cited include: (1) Class 1 piping between 2- and 4-inch nominal pipe size (NPS), (2) Class 2 piping less than 4-inch NPS and (3) Class 2 piping greater than 4-inch NPS, but less than 3/8-inch in wall thickness. For the first example, SNC states that the RI-ISI program will now require volumetric examination. However, for examples 2 and 3, SNC simply states that the RI-ISI program will now require examination. Please, clarify the type of examination (volumetric, surface or visual) that will be applied to these new inspection elements as a result of the RI-ISI process. It should be noted that, if the new examinations are simply visual VT-2, the current ASME Code program contains this requirement; therefore, no new examinations are being implemented.

Response

The examination type for all the examples cited in Section 5 of our July 17, 2003 submittal is "volumetric."

Question 6

It is noted that Tables 5-1a and 5-1b of the WCAP are intended to summarize and compare new RI-ISI with existing ASME Code examinations, list the relevant degradation mechanisms for elements (examination locations) by plant system, and include other relevant information. There are several questions related to the information contained in this Table, as follows:

a) In order to determine if appropriate examinations methods are being correctly applied to target specific degradation, further clarification is necessary. Please "break-out" the planned methods for examination (i.e., show how many volumetric or surface examinations will be applied as a result of the RI-ISI process, instead of listing these only as "NDE [non-destructive examination])."

b) Similarly, describe the type of visual examinations that will be applied for those components where "VT" is listed. Since footnote (a), for Tables 5-1a and 5-1b of the WCAP, specify that VT-2 examinations during system pressure tests will continue to be performed per ASME Code requirements, differentiate between any VT-2 examinations performed as a result of the RI-ISI process and how any new visual VT-2 examinations provide an adequate margin of safety, since they may already be required by the ASME Code. Also, identify if any VT-1 or VT-3 examinations are being applied to the inside surfaces of the subject piping.

c) Identify the type and extent, if any, of the listed "NDE" (volumetric or surface) examinations that are being performed to satisfy existing augmented inspection programs versus being the result of RI-ISI process evaluations.

d) Under the Table column "Degradation Mechanism(s)," it is unclear which mechanisms go with which ASME Code Category welds. Several mechanism designations (MF [Mechanical Fatigue], TF [Thermal Fatigue], SCC [Stress Corrosion Cracking], VF [Vibrational Fatigue]) are multiply listed for several plant systems. Please clarify how to interpret the information in this column.

Response

a) Table 5-1a and Table 5-1b (Attachment 1) have been revised to break-out the planned NDE methods into volumetric or surface (penetrant examinations) examinations.

b) At this time, there are not any VT-1 or VT-3 examinations being applied to the inside surfaces of the subject piping; therefore, all visual examinations listed Table 5-1a and Table 5-1b refer to "VT-2" visual examinations. Also, there are no new VT-2 examinations proposed. The VT-2 examinations listed in Table 5-1a and Table 5-1b pertain to the examination of socket-welded segments in the RCS system and these examinations are the same VT-2 examinations as currently being conducted per ASME Section XI.

VT-2 examinations are required by Table 4.1-1 of the WCAP for Item No. R1.12, "Elements Subject to High Cycle Mechanical Fatigue" for socket-weld segments experiencing vibration; however, Item No. R1.11, "Elements Subject to Thermal Fatigue" requires a volumetric examination of all welds. However, volumetric examination of socket welds and branch connections 2-inch (nominal pipe size) NPS and smaller is inconclusive due to the geometric limitations imposed by the socket welds and the small connection size. Attempted compliance with the WCAP volumetric requirements will result in unusual difficulty without a compensating increase in the level of quality and safety. Other utilities such as Duke and Dominion have submitted relief requests and received NRC staff approval to perform VT-2 examinations in lieu of the required volumetric examinations. SNC proposes a deviation (See Attachment 2) to the WCAP requirements for socket weld examinations and branch connection examinations (2-inch NPS and smaller) based on the NRC's safety evaluation dated September 23, 2003 for Surry. In lieu of the WCAP required volumetric examination, a VT-2 or surface examination will be performed as described in Table 1 of the deviation.

c) Table 5-1a and Table 5-1b notes (d, e, h, l, and m) have been added and/or modified to assist in identifying the type and extent of examinations that are being performed to satisfy existing augmented inspection programs versus those being examined as a result of RI-ISI process.

d) Table 5-1a and Table 5-1b have been modified to separate out "Degradation Mechanisms" to correspond with the appropriate ASME Code Category, to the extent practical. For those cases where there is more than one set of degradation mechanisms refer to Note k.

Question 7

Observation SY-02 – Point 5, and SY-07 from the peer review of the Farley Probabilistic Risk Assessment (PRA) indicate that there appeared to be no common cause failures (CCF's) modeled between redundant trains with some pumps in standby and others operating (service water and component cooling pumps respectively). SNC states that these CCFs will be modeled in a future revision. Please explain why the lack of these CCFs models in Revision 5 are not expected to affect the RI-ISI conclusion or otherwise evaluate the potential impact. For example, Observation DA-05 also discusses apparent CCF modeling weakness but SNC provides a reasonable argument that the diesels' CCF values will have little to no impact on the pipe rupture events that dominate the RI-ISI evaluation.

Response

For the LOCA scenarios that dominate the RI-ISI evaluation, those pumps in the Service Water (SW) and Component Cooling Water (CCW) systems that are running at the start of the event will remain running. Therefore, changes to the modeling of common cause failure-to-start between the running and standby pumps are expected to have little impact on the RI-ISI evaluation.

For the SW system, four of the five pumps are normally running and will remain running following a LOCA. The success criterion for the SW system, following a LOCA, requires two pumps per train. Therefore, the major common cause failure concern affecting the RI-ISI is common cause failure-to-remain-running of the two pumps in operation prior to the event. This is included in the modeling. Common cause failure-to-run between the standby and running SW pumps is not considered as likely as common cause failure between the pumps running prior to the event because there is a significant difference in run-hours between the train-dedicated SW pumps and the swing, standby pump (the swing pump accumulates approximately 50% of the running hours accumulated by each of the train-dedicated pumps on an annual basis).

For the CCW system, the run-hours between the train-dedicated and swing pumps are nearly equal on an annual basis. Therefore, common cause failure-to-remain-running of all three CCW pumps was included in the Revision 5 PRA model used for the RI-ISI evaluation.

Since the dominant common cause events for the scenarios of interest in the RI-ISI are included in the Revision 5 model, the comments in the peer review observation and any future model revisions to address these comments are not expected to change the conclusions reached in the RI-ISI evaluation.

Question 8

Observation DA-02 notes that there are significant differences between the CCF values used in the Farley PRA (based on CCF estimates developed in the 1990s) and current generic values. Performance of the WCAP uncertainty analysis will not correct for large and potentially inappropriate deviations in mean values. Please identify SNCs current CCF estimates that vary significantly from the generic estimates and verify your estimates using the current methodologies and generic estimates, or explain why your values are not expected to affect the conclusions of the RI-ISI submittal.

Response

The subject peer review observation documents a subjective judgment on the part of the reviewer that there would be significant differences between the unscreened values in NUREG/CR-5497 and the CCF values used in the current Farley analysis. However, the reviewer could not cite a specific example of where this was the case. Use of the CCF data from NUREG/CR-5497 without performance of plant-specific screening to ensure that the events are applicable to the design and operation of Farley Nuclear Plant would not be a valid application of common cause methods. Therefore, a comparison of the current CCF values used in the Farley PRA model with the generic values from NUREG/CR-5497 would provide no indication of how the RI-ISI results might change with future application of the updated CCF database.

Any changes which would occur in the plant-specific CCF factors as a result of implementing the NUREG/CR-5497 database would affect all segments equally. Therefore, the relative risk ranking results between segments used in the Westinghouse RI-ISI methodology would not be expected to be impacted.

Question 9

Observation HR-04 and HR-05 relate to the lack of calibration error modeling and identify other questionable details (i.e., the use of a 0.1 multiplier) in the human error analyses used in the PRA. The NRC staff's safety evaluation report on the Individual Plant Examination also noted that calibration errors were not included in the models and questioned the limited and probably optimistic treatment of diagnosis and the "blanket" application of selected performance shaping factors (PFS) without case-by-case assessment. Part of your response on calibration errors is that the miscalibration errors are included in the reliability and common cause failures. However, as noted in the previous two questions, the NRC staff has some reservations with SNCs common cause analyses. Although your submittal indicates that SNC has individually reviewed human errors such that blanked application of PFS may no longer be a concern, the continued use of the multiplicative factors indicate that the human error analyses may not yet be complete and your response to Observation HR-05 indicates that you continue to review the human reliability analysis and will update the models and values as appropriate. Performance of the WCAP uncertainty analysis will not correct for large and potentially inappropriate deviations in mean values. Please, explain why these difficulties associated with the human error analysis are not expected to affect the RI-ISI conclusion, or otherwise evaluate the potential weakness.

Response

As noted in your question, CCF type calibration errors were accounted for in the Farley CCF analysis. Thus, CCF events include contributions from CCF type calibration errors. The current treatment has been applied consistently and captures important miscalibration errors in the equipment failure probabilities. The separation of the current basic events representing failure of components subject to miscalibration failure modes into separate events for miscalibration and hardware failures would not affect the conclusions of the RI-ISI evaluation.

The use of a 0.1 multiplier applies only to those human error events analyzed using the THERP methodology. In the Farley PRA model, THERP is used for pre-initiating event human errors and for limited recovery events. The major human error events for operator response to initiating events using the Westinghouse Emergency Response Guidelines such as alignment of Emergency Core Cooling System recirculation, were evaluated using the Success Likelihood Index Method

(SLIM). The human error probabilities for these major operator responses to LOCA events have been compared with those used by other Westinghouse Owners Group (WOG) plants using the WOG PRA Comparison database. In addition, the values used for these operator actions have been evaluated against the Checklist for Technical Consistency in a PSA Model contained in the EPRI PSA Applications Guide (TR-105396), and have been reviewed as part of the NRC benchmarking effort for the Significance Determination Process. No significant differences have been identified in these comparisons. Therefore, this issue is not expected to impact the evaluation of the LOCA events primarily used in the RI-ISI methodology.

As noted above for the CCF analysis, any changes in the human error analysis will equally affect all of the analyzed piping segments. Since the risk ranking is based on a relative ranking of the individual piping contributions to risk, resolution of the issues raised in this observation is not expected to change the conclusions of the RI-ISI evaluation.

ATTACHMENTS

ATTACHMENT 1 - REVISED TABLES 5-1a AND 5-1b

ATTACHMENT 2 - SOCKET WELD EXAMINATION DEVIATION

Attachment 1

Table 5-1a (Revised)
FARLEY UNIT 1 STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS

System	No. of HSS Segments (No. of HSS in Aug Program / Total No. of Segments in Aug Program)	Degradation Mechanism (k)	Class	ASME Code Category	Weld Count		ASME Section XI Exam Methods		RI-ISI[a]	
					Butt	Socket	Vol & Surface[j]	Surface Only[j]	SES Matrix Region	Number of Exam Locations
1AFW	3(0/0)	MF, TF	2	C-F-2	18	0	2	0	2	3 VOL
1CHI	0	MF, TF	2	(b)	(b)	(b)	(b)	(b)	-	0
1CI	0	MF, TF & MF, TF, E/C	2	(b)	(b)	(b)	(b)	(b)	-	0
1CS	0	MF, TF & MF, TF, VF	2	C-F-1	132	0	10	0	-	0
1CV	0	MF, TF, VF	1	B-J	3	38	0	11	-	0
	0	MF, TF & MF, TF, VF	2	C-F-1	415	146	32	11	2	2 ΔRisk[g] VOL
	3(0/0)	MF, TF	2	(c)	(c)	(c)	(c)	(c)	2	3 VOL
1FW	3(3/6)[e]	MF, TF, E/C	2	C-F-2	63	0	5	0	1A, 1B	3 VOL
1MS	9(6/18)[d]	MF, TF, VF	2	C-F-2	153	0	12	0	1A, 2	49 VOL
	5(0/0)[l]	MF, TF, VF	2	(c)	(c)	(c)	(c)	(c)	2	5 VOL

Attachment 1

Table 5-1a (Revised)
FARLEY UNIT 1 STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS

System	No. of HSS Segments (No. of HSS in Aug Program / Total No. of Segments in Aug Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count		ASME Section XI Exam Methods		RI-ISI[a]	
					Butt	Socket	Vol & Surface[j]	Surface Only[j]	SES Matrix Region	Number of Exam Locations
1RC	15	MF, TF, SCC& MF, TF, SCC, VF	1	B-F	18	0	18	0	1A, 2	18 VOL
	15	MF, TF & MF, TF, VF	1	B-J [f]	456	206	114	52	1B, 2	10 VOL +5 PT[n]+ 1 VT-2[i]
	18	MF, TF	1	(c)	(c)	(c)	(c)	(c)	2	17 VT-2[i]+1 PT[n]
1RH	18	MF, TF& MF, TF, VF	2	C-F-1	380	0	29	0	2	18 VOL
1SGB	0(0/3) [h]	MF, TF& MF, TF, E/C	2	(b)	(b)	(b)	(b)	(b)	-	0
1SI	7	MF, TF	2	C-F-1	413	267	31	21	2	7 VOL
1SS	0	MF, TF	2	(b)	(b)	(b)	(b)	(b)	-	0
Total	96 (9 / 27)		1	B-F	18	0	18	0	1A, 2	18 VOL
				B-J	459	244	114	63	1B, 2	10 VOL + 1 VT-2[i]+ 5 PT
				(c)	(c)	(c)	(c)	(c)	2	17 VT-2[i]+ 1 PT
			2	C-F-1	1340	413	102	32	2	27 VOL
				C-F-2	234	0	19 [d]	0	1A, 1B, 2	55 VOL
				(b)	(b)	(b)	(b)	(b)	-	0
				(c)	(c)	(c)	(c)	(c)	2	8 VOL
			Total		2051	657	253	95	1A, 1B, 2	118 VOL +18 VT-2+6 PT

Attachment 1

Table 5-1b (Revised)
FARLEY UNIT 2
STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS

System	No. of HSS Segments (No. of HSS in Aug Program / Total No. of Segments in Aug Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count		ASME Section XI Exam Methods		RI-ISI[a]	
					Butt	Socket	Volumetric & Surface[j]	Surface Only[j]	SES Matrix Region	Number of Exam Locations
2AFW	3(0/0)	MF, TF	2	C-F-2	21	0	2	0	2	3 VOL
2CHI	0	MF, TF	2	(b)	(b)	(b)	(b)	(b)	-	0
2CI	0	MF, TF & MF, TF, E/C	2	(b)	(b)	(b)	(b)	(b)	-	0
2CS	0	MF, TF & MF, TF, VF	2	C-F-1	139	0	11	0	-	0
2CV	0	MF, TF, VF	1	B-J	3	30	0	9	-	0
	0	MF, TF & MF, TF, VF	2	C-F-1	400	94	30	8	-	0
	3 (0/0)	MF, TF	2	(c)	(c)	(c)	(c)	(c)	2	3 VOL
2FW	3(3/6) [e]	MF, TF, E/C	2	C-F-2	60	0	5	0	1A, 1B	3 VOL
2MS	9(6/18) [d]	MF, TF, VF	2	C-F-2	146	0	11	0	1A, 2	42 VOL
	5(0/0) [m]	MF, TF, VF	(c)	(c)	(c)	(c)	(c)	(c)	1A	25 VOL

Attachment 1

Table 5-1b (Revised)
FARLEY UNIT 2
STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS

System	No. of HSS Segments (No. of HSS in Aug Program / Total No. of Segments in Aug Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count		ASME Section XI Exam Methods		RI-ISI[a]	
					Butt	Socket	Volumetric & Surface[j]	Surface Only[j]	SES Matrix Region	Number of Exam Locations
2RC	15	MF, TF, SCC& MF, TF, SCC, VF	1	B-F	18	0	18	0	1A, 2	18 VOL
	15	MF, TF & MF, TF, VF	1	B-J [f]	467	186	117	47	1B, 2	10 VOL +5 PT[n]+ 1 VT-2[i]
	18	MF, TF	1	(c)	(c)	(c)	(c)	(c)	2	17 VT-2[i]+1 PT[n]
2RH	19	MF, TF& MF, TF, VF	2	C-F-1	369	0	28	0	2	19 VOL
2SGB	0 (0/3) [e]	MF, TF & MF, TF, E/C	2	(b)	(b)	(b)	(b)	(b)	-	0
2SI	8	MF, TF	2	C-F-1	388	136	30	11	2	8 VOL
2SS	0	MF, TF	2	(b)	(b)	(b)	(b)	(b)	-	0
TOTAL	98 (9 / 27)		1	B-F	18	0	18	0	1A, 2	18 VOL
				B-J	470	216	117	56	1B, 2	10 VOL + 1 VT-2[i]+ 5 PT
				(c)	(c)	(c)	(c)	(c)	2	17 VT-2[i]+1 PT[n]
			2	C-F-1	1296	230	99	19	2	27 VOL
				C-F-2	227	0	18	0	1A, 1B, 2	48 VOL
				(b)	(b)	(b)	(b)	(b)	-	0
				(c)	(c)	(c)	(c)	(c)	1A, 2	28 VOL

Attachment 1

**Table 5-1b (Revised)
FARLEY UNIT 2
STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS**

System	No. of HSS Segments (No. of HSS in Aug Program / Total No. of Segments in Aug Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count		ASME Section XI Exam Methods		RI-ISI[a]	
					Butt	Socket	Volumetric & Surface[j]	Surface Only[j]	SES Matrix Region	Number of Exam Locations
			Total		2011	446	252	75	1A, 1B, 2	131 VOL+18 VT-2 + 6 PT

**Table 5-1a and b (Revised)
FARLEY UNIT 1 and 2
STRUCTURAL ELEMENT SELECTION
NOTES**

Summary:

Current ASME Section XI selects a total of 348 non-destructive exams for FNP Unit 1 while the RI-ISI program selects a total of 142 exams (118 volumetric, 18 VT-2, and 6 PT), which results in a 59% reduction. Without inclusion of the augmented Technical Specification required examinations, there would be a total of 102 RI-ISI exams, which results in a 71% reduction.

Current ASME Section XI selects a total of 327 non-destructive exams for FNP Unit 2 while the RI-ISI program selects a total of 155 exams (131 volumetric, 18 VT-2, and 6 PT), which results in a 53% reduction. Without inclusion of the augmented Technical Specification required examinations, there would be a total of 122 RI-ISI exams, which results in a 63% reduction.

Degradation Mechanisms: VF – Vibratory Fatigue; TF – Thermal Fatigue; MF – Mechanical Fatigue; E/C – Erosion/Corrosion; SCC – Stress Corrosion Cracking

Notes for Table 5-1a and Table 5-1b:

Attachment 1

a. System pressure test requirements and VT-2 visual examinations shall continue to be performed in all ASME Code Class 1 and 2 systems.
b. Piping is exempt per the requirements of the 1989 Edition of Section XI.
c. Piping in this grouping is exempt per the requirements of the 1989 Edition of Section XI; however, examinations are required per the proposed RI-ISI program.
d. There are eighteen Class 2 Main Steam System segments in the high-energy Technical Specification augmented weld inspection program. Twelve of the eighteen segments are designated as Low Safety Significant (LSS) and will continue to be examined per Technical Specification requirements. The six remaining segments are designated as Highly Safety Significant (HSS) (SES Region 1A) with forty-six Farley-1 welds (thirty-nine Farley 2 welds) volumetrically examined (on the six HSS segments) per Technical Specification Requirements. (If these segments were not augmented, one weld on each segment, or six welds would be examined for RI-ISI). There are three additional HSS segments (SES Region 2) located inside containment, outside of scope of the augmented weld inspection program, with one weld on each segment volumetrically examined. In conclusion, there would be nine Class 2, Main Steam, C-F-2 welds selected per Farley unit for RI-ISI if there was not an augmented program.
e. There are six Class 2 Feedwater System segments in the flow-accelerated corrosion (FAC) program [also known as erosion/corrosion or E/C)]. Three of the segments are designated as LSS and thickness examinations will continue to be performed per the FAC program. The remaining three segments are designated as HSS segments (SES Region 1a, 2). Beyond the continuing FAC thickness examinations which satisfy the Region 1A examination requirements, one weld on each of the three HSS segments will be volumetrically examined for cracking to satisfy the Region 2 examination requirements.
f. Monitoring program continues for high-cycle thermal fatigue (stratification/cycling) issues.
g. Two volumetric weld examinations added for change in risk considerations for FNP-1. None were added for FNP-2.
h. There are three Class 2 Steam Generator Blowdown segments in the flow-accelerated corrosion (FAC) program. There are no HSS segments; therefore, the only examinations are the augmented FAC thickness examinations.
i. VT-2 examination once per refueling outage to detect ID initiated through-wall flaws because volumetric examination is impractical.
j. Includes only those weld examinations required to meet the requirements of the 1989 Edition of Section XI.
k. When there is more than one set of failure mechanisms shown on a row, segments on that row have different failure mechanisms. For example, the majority of the Class 2, Category C-F-2 segments in CS are subject to mechanical and thermal fatigue and these segments are listed "MF, TF." However, some of the Class 2, Category C-F-2 segments in CS are also subject to vibratory fatigue and these segments are listed as "MF, TF, VF."
l. There are five HSS segments previously exempt from Section XI examination requirements. These segments are not in the high-energy Technical Specification augmented weld inspection program, thus one weld will be volumetrically examined on each segment.
m. There are five HSS segments previously exempt from Section XI examination requirements. These segments are not in the high-energy Technical Specification augmented weld inspection program; however, they are SES Region 1A (because they are HSS and have a high failure importance due to vibration). Since they are SES Region 1a, all butt welds will be volumetrically examined on each segment (total of 25 welds) to meet RI-ISI requirements.
n. Surface examination performed once per ten years.

Attachment 2

Deviation to the WCAP

WCAP-14572, A-version, Table 4.1-1, Category R-A, Item R1.12 requires a VT-2 examination for elements subject to high cycle mechanical fatigue (e.g., HSS socket-welded segment RC-026C is subject to vibration because it is near a reactor coolant pump); however, Item R1.11 requires a volumetric examination of segments selected for examination that are subject to thermal fatigue. This requirement encompasses the examination of socket welds as well as butt welds, and the use of non-volumetric examination methods for Item R1.11 socket-welded segments is a deviation to the requirements of WCAP-14572, A-version. VT-2 visual examinations and surface examinations, as applicable, are planned for use at Farley for Item R1.11 socket-welded segments. The basis for this deviation from the prescribed examination method is given below. (See Table 1 for a list of the affected HSS socket-welded segments).

Basis for the Deviation

There have been numerous EPRI reports developed pertaining to the failure of socket welds. Examples are, TR-104534, "Fatigue Management Guideline," TR-107455, "Vibration Fatigue of Small Bore Socket-Welded Pipe Joints," TR-113890, "Vibration Fatigue Testing of Socket Welds (PWRMRP-07)," and TR-111188, "Vibration Fatigue Testing of Socket Welds." These documents indicate that the majority of fatigue failures of socket welds are caused by vibration and the majority of the failures occurred due to cracks that initiated at weld roots.

Volumetric examination of socket welds and associated branch connection welds (that connect the socket-welded line to a larger line) to detect a flaw originating at the weld root is inconclusive and impractical due to the geometric configuration of the welded joint. The ASME Section XI Code recognized this issue and thus the existing Code does not require volumetric examination of socket welds, but instead requires surface examination. However, there is not any benefit in performing surface examinations to detect cracks initiating at the weld root. By the time that a crack initiating at the weld root is detectable with surface examination methods, there would be through wall leakage. Therefore, for flaws originating at the inside diameter (ID) of the pipe, a visual examination (VT-2) for through-wall leakage each refueling outage (per ASME Section XI) is the most appropriate examination method.

While most failures of socket-welded segments occur due to ID initiated cracks, it is recognized that surface examinations may be an effective method for discovery of certain potential surface flaws originating on the outside diameter (OD) of the pipe. Specifically, these are: (1) OD flaws induced by external chloride stress corrosion cracking and (2) OD flaws induced by low-cycle, high-bending stress thermal fatigue. The Farley configurations are addressed below for these two OD cracking cases:

1. Drains, Vents, Test Connections, Capped Lines, Instrument Lines - Material controls and cleanliness requirements at Farley have prevented and should continue to prevent OD initiated external chloride stress corrosion cracking on these socket-welded segments. Flaws induced by low-cycle, high-bending stress thermal fatigue should not occur on these segments because these configurations are short and are not subject to out of design conditions that could cause high-bending stresses. Since OD cracking is not expected, surface examinations are not beneficial and a visual examination (VT-2) for through-wall leakage once per refueling outage, per ASME Section XI, is the most appropriate examination method.

Attachment 2

2. Excess Letdown Line, Safety Injection Lines, and Drain Tank Lines – Material controls and cleanliness requirements at Farley have prevented and should continue to prevent OD initiated external chloride stress corrosion cracking on these socket-welded segments. However, because of the routing, there remains a limited potential for low-cycle, high-bending stress thermal fatigue in these segments. Since there is a slight potential for OD cracking, a surface examination will be performed on one weld per segment (each 10-year interval).

This deviation is consistent with the proposed ASME Section XI Code "Appendix X, Risk-Informed Requirements for Piping" for Item R1.11 socket-welded segments and branch piping connection welds (NPS 2 and smaller). Appendix X replaces the volumetric examination with a visual examination (VT-2), while the proposed deviation replaces the volumetric examination with a visual examination (VT-2) (plus surface examinations, as appropriate). [The proposed Appendix X has been approved by ASME Subgroup - Water Cooled Systems. This subgroup has responsibility for ASME Code actions related to the implementation of risk-informed ISI. The appendix is presently with the ASME Section XI Subcommittee for review and approval.]

Conclusion

The proposed alternative examinations listed in Table 1 will provide reasonable assurance of structural integrity. Attempted compliance with the WCAP volumetric requirements will result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the use of the alternate examinations in lieu of the volumetric examinations should be approved.

Attachment 2

Table 1 Farley Risk-Informed ISI HSS Socket-Welded Segments for Proposed Deviation				
Segments	**NPS**	**Description**	**RI-ISI Exam**	**Comments**
RC-012A, B RC-020A, B, C	3/8"-3/4" 3/8"-3/4"	Instrument Lines	VT-2	Not subject to OD cracking. The RI-ISI exam is the same as the Section XI required VT-2 exam.
RC-054	1"	Thermowell	VT-2	Not subject to OD cracking. The RI-ISI exam is the same as the Section XI required VT-2 exam.
RC-017	1"	Capped Lines	VT-2	Not subject to OD cracking. The RI-ISI exam is the same as the Section XI required VT-2 exam.
RC-018A, B, C	1"	RTDs	VT-2	Not subject to OD cracking. The RI-ISI exam is the same as the Section XI required VT-2 exam.
RC-032A, B, C RC-036A, B, C RC-041A	3/4"	Vents, Drains, and Test Connections	VT-2	Not subject to OD cracking. The RI-ISI exam is the same as the Section XI required VT-2 exam.
RC-024B, C	2"	RC Drain to Drain Tank	Surface	Potentially subject to OD cracking. The RI-ISI exam is the same as the Section XI required surface exam.
RC-029A, B, C	2"	Safety Injection Lines	Surface	Potentially subject to OD cracking. The RI-ISI exam is the same as the Section XI required surface exam.
RC-059	1"	Excess Letdown Line	Surface	Potentially subject to OD cracking. The RI-ISI exam is in excess of the Section XI required VT-2 exam.