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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Units 1 and 2
Proposed Risk-Informed Inservice Inspection Program
Request for Additional Information (TAC Nos. MD3137 and MD3138)

References:

1. Letter from J. N. Jensen, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Request for Approval of Risk-Informed Inservice Inspection Program for Class 1 and 2 Piping American Society of Mechanical Engineers Code, Category B-F, B-J, C-F-1, and C-F-2 Piping Welds," AEP:NRC:6055-09, Accession Number ML062850540, dated September 29, 2006.

2. Electronic Communication from P. S. Tam, NRC, to M. K. Scarpello, I&M, "Draft RAI on D. C. Cook Risk-Informed ISI Program (TAC Nos. MD3137, 8)," Accession Number ML070890463, dated March 29, 2007.

3. Electronic Communication from P. S. Tam, NRC, to M. K. Scarpello, I&M, "D. C. Cook – Draft RAI Questions re: Risk-Informed ISI Program (TAC Nos. MD3137, 8)," Accession Number ML070990628, dated April 9, 2007.

By letter dated September 29, 2006 (Reference 1), Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant Units 1 and 2, proposed an alternative to the American Society of Mechanical Engineers (ASME) Code Section XI. Specifically, I&M proposed adopting a risk-informed inservice inspection program using ASME Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI, Division 1."

In References 2 and 3, the Nuclear Regulatory Commission (NRC) requested additional information regarding I&M's proposed alternative. The attachment to this letter provides I&M's response to the NRC's requests for additional information.

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This letter contains no new commitments. Should you have any questions, please contact Ms. Susan D. Simpson, Regulatory Affairs Manager, at (269) 466-2428.

Sincerely,

Joseph N. Jensen
Site Vice President

Attachment: Risk-Informed Inservice Inspection Program, Request for Additional Information

c: R. Aben – Department of Labor and Economic Growth
J. L. Caldwell – NRC Region III
K. D. Curry – AEP Ft. Wayne, w/o attachment
J. T. King – MPSC
MDEQ – WHMD/RPMWS
NRC Resident Inspector
P.S. Tam – NRC Washington, DC

Risk-Informed Inservice Inspection Program
Request for Additional Information

By letter dated September 29, 2006 (Reference 1), Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposed an alternative to the American Society of Mechanical Engineers (ASME) Code Section XI. Specifically, I&M proposed adopting a risk-informed inservice inspection program using ASME Code Case N-716 (N-716), "Alternative Piping Classification and Examination Requirements, Section XI, Division 1."

In References 2 and 3, the Nuclear Regulatory Commission (NRC) requested additional information regarding I&M's proposed alternative. The following provides I&M's response to the NRC's requests for additional information (RAIs).

NRC March 29, 2007, RAI (Reference 2)

NRC Request 1

The licensee requests authorization to implement a risk-informed inservice inspection (ISI) program based on American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Case N-716 (N-716). There appears to be, however, some differences between the methodology in N-716 and the method applied by the licensee:

a. *Table 3 of N-716 discusses high, medium, and low failure potential, and pairs these potentials with degradation categories large break, small leak, and none respectively. It does not appear that this table was used in the submittal. Was this table used in the submittal? If not, what was used in lieu of Table 3?*

   I&M Response to Request 1.a

   The information contained in Table 3 of N-716 was used in the Reference 1 proposed alternative. The information is identified in Reference 1, Tables 3.4-1, 3.4-2, 5-1, and 5-2 in the column labeled "Failure Potential." This column is further divided into two sub-columns (i.e., DMs (degradation mechanisms) and Rank). The Failure Potential Rank for high safety significant (HSS) locations is then assigned as high, medium, or low, depending upon potential susceptibly to the various types of degradation. Low safety significant (LSS) locations were conservatively assumed to be a rank of medium (see response to NRC Request 4.b).

b. *Section 5(c) of N-716 does not appear to provide a "with probability of detection (POD)" and "without POD" option in the calculation, but the submittal includes one set of estimates for "with POD" and another "w/o POD" in Table 3.4-1. Please*

clarify how the "with POD" and "w/o POD" columns in Table 3.4-1 are consistent with Section 5(c) of N-716.

I&M Response to Request 1.b

It is true that N-716 does not discuss the two options presented above. Reference 1 contained both options in order to be consistent with previous risk-informed inservice inspection (RI-ISI) submittals that contained both options. These two sets of analyses are typically conducted to provide a sensitivity of the change-in-risk evaluation with respect to assumptions on POD.

c. *The estimates in the "w/o POD" column in Table 3.4-1 all seem to include a standard POD of 0.5. Is this correct? If not, please provide some examples using the conditional core damage probability (CCDP) values from page 11 of 35 to produce the entries in Table 3.4-1 and 3.4-2.*

I&M Response to Request 1.c

The w/o POD column applies a POD of 0.5 for both the ASME Code Section XI program, and the N-716 program. Thus, there is no extra credit assumed for an N-716 inspection as compared to ASME Code Section XI inspection, as to inspection effectiveness (e.g., due to larger inspection volumes in the N-716 program).

d. *Section 7 of N-716, "Program Updates," includes several steps that make up a program update. Page 15 of 35 in the licensee's submittal states that, "[u]pon approval of the RI_B Program, procedures that comply with the guidelines described in Reference 2 [Electric Power Research Institute (EPRI) TR-112657 (EPRI Topical)] will be prepared to implement and monitor the program." Please identify the Sections in the EPRI topical that describe the update program that the licensee intends to implement. Please describe and compare the update program that the licensee intends to implement against the characteristics of such a program as described in Section 7 of N-716.*

I&M Response to Request 1.d

The wording in Reference 1 was based on previous RI-ISI submittals. While the intent of both updating processes (EPRI TR-112657 and N-716) is the same, I&M will meet the wording of N-716.

NRC Request 2

Regulatory Guide (RG) 1.178, "An Approach for Plant-Specific Risk-Informed Decision making for Inservice Inspection of Piping," describes one acceptable process for developing an RI-ISI program. Please explain how:

a. The approach used to analyze piping system failures for the plant-specific PRA [probabilistic risk assessment] of pressure boundary failures compares to the approach described in Section 2.1.4 of RG 1.178;

I&M Response to Request 2.a

The purpose of segments and segment definitions is identical between the N-716 approach and that of the EPRI RI-ISI methodology. In both methodologies, segments are used only as an accounting/tracking tool. That is, whether the weld is tracked individually or as part of a segment, the results of the risk ranking and element selection part of the methodology will not change. In both approaches, whether the segment is small (e.g., a single weld) or large (e.g., many welds), all of the welds will be ranked and then subject to a fixed sampling percentage for determining the size of the inspection population.

As an example, if the population of HSS welds is 100, whether they are tracked as ten segments (e.g., ten welds per segment) or two segments (50 welds per segment), all 100 welds would be subject to the element selection process. For example, 25 percent (%) of HSS welds with susceptibly to a degradation mechanism would be selected for N-716 applications and 25% of welds identified as Risk Category 2 would be selected for EPRI RI-ISI applications.

b. The process used to assess piping failure potential for the plant-specific PRA of pressure boundary failures compares to the process outlined in section 2.1.5 of RG 1.178;

I&M Response to Request 2.b

For N-716 applications, failure potential is used in two ways:

To confirm on a plant-specific basis that there is no other piping that should be considered as HSS per Section 2(a) of N-716 (see responses to NRC Requests 2.c and 6.c).

Once the HSS population has been determined for the plant, the failure potential evaluation is identical to that in EPRI TR-112657 as applied to a number of approved RI-ISI applications. That is, the degradation mechanisms assessed, the

evaluation criteria (e.g., attributes such as operating temperatures, allowable temperature differentials, susceptible materials, flow velocities, etc.), and the failure potential ranking are the same.

c. *The quantitative results of the pipe failure frequency that resulted from the failure potential assessment compares to the weld failure frequencies proposed in Section 5(a) of N-716 that are eventually used in your change-in-risk estimates;*

I&M Response to Request 2.c

Because the failure frequencies in Section 5(a) of N-716 are at the weld level, they are substantially smaller than what is used in conducting an internal flooding study in general, and the CNP internal flooding study in particular (see response to NRC Request 6.c). Another reason the failure frequencies used in the CNP internal flooding study are larger than the values used in the N-716 application is that the CNP internal flooding study includes the impact of flood sources beyond piping (e.g., tanks, pumps, heat exchangers, etc.). For screening purposes, this is conservative from an internal flooding study perspective.

d. *The consequence evaluation performed as part of the plant-specific PRA of pressure boundary failures compares with the process outlined under Section 2.1.6 of RG 1.178.*

I&M Response to Request 2.d

The plant-specific PRA of pressure boundary failures is consistent with that discussed in Section 2.1.6 of RG 1.178 in that plant walkdowns were conducted to identify flood initiators and the locations of critical components. Additionally, for each flood zone and/or scenario, the impact of both direct and indirect effects was considered. Direct effects included loss of a train or system (e.g., loss or diversion of flow), an initiating event, or both. Indirect effects included spatial effects such as spray, pipe whip, etc., as well as loss of inventory effects (e.g., loss of a common tank).

NRC Request 3

Please fully define the population of welds to which the 10% guideline is applied and what inspections are counted.

a. *Is the guideline to examine a minimum 10% of all high-safety-significant (HSS) welds, 10% of all HSS butt welds, 10% of all HSS butt welds [greater than or equal to 4-inch nominal pipe size (>= 4 NPS),] or something else?*

I&M Response to Request 3.a

The guideline is to examine a minimum of 10% of HSS welds. For CNP, this population includes welds that are less than, equal to, and greater than 4 NPS. It also includes butt welds and socket welds.

Additionally, a lesson learned from the CNP proposed-alternative submittal (Reference 1) was that the wording of N-716 could be clearer in its intent to require inspection of at least 10% of the reactor coolant pressure boundary (RCPB) welds. While the CNP proposed-alternative submittal meets this intent, it is I&M's understanding that N-716 will be revised to make this requirement clearer and to reflect other lessons learned from N-716 applications (see response to NRC Request 4.a).

b. *What type of inspections can be counted, e.g., can visual examinations or wall thickness exams be counted in the 10%?*

I&M Response to Request 3.b

Per N-716, wall thickness exams as part of the flow accelerated corrosion (FAC) and localized corrosion (excluding crevice corrosion) programs cannot be counted as part of the 10% required population. Because of the nature of the degradation, wall-thinning examinations for locations potentially susceptible to erosion-cavitation will be conducted.

Per N-716, the requirements for examination of socket welds and smaller bore branch connections (i.e., less than 2 NPS) susceptible to thermal fatigue shall be a volumetric examination of the piping base metal within 1/2-inch of the toe of the weld and a visual examination of the fitting itself. This is consistent with the requirements of EPRI MRP-146, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines." The MRP-146 evaluation has shown no small bore piping susceptible to swirl penetration thermal fatigue at CNP.

Thus, HSS inspections required by N-716 shall be volumetric examinations as part of the CNP proposed alternative.

c. *What percentage of Class 1 butt welds (regardless of [nominal] pipe size (NPS)) will be inspected in the proposed risk-informed program?*

I&M Response to Request 3.c

I&M has selected a 14.7% sample for Unit 1 and a 10.1% sample for Unit 2 of Class 1 butt welds for examination, regardless of NPS.

NRC Request 4

Section 5(c) in N-716 does not clearly specify what population of welds should be included in the change of risk estimates and what welds may be excluded. The description of the parameters in the equations in Section 5(c) indicates that any weld that was inspected under Section XI or that will be inspected under the RI-ISI program will be included in the change-in-risk estimate.

a. *Is the population of welds that should be included in the N-716 change-in-risk estimate all welds that were inspected under Section XI, and that will be inspected under the RI-ISI program? If not, where in Code Case N-716 is the guidance that reduces the population of welds that should be included in the change-in-risk estimate.*

I&M Response to Request 4.a

The population of welds that needs to be included in the change-in-risk assessment includes all welds receiving nondestructive examination (NDE) except for those that receive only a surface examination and are not susceptible to outside diameter attack (e.g., external chloride stress corrosion cracking). This population includes so-called "risk category 6 and 7" locations, which are not required to be included in the RI-ISI change-in-risk assessment.

It is I&M's understanding that N-716 will be updated to reflect this requirement (i.e., exclusion of surface only examinations without outside diameter attack), as well as any other relevant feedback from N-716 applications.

b. *If all welds that were or will be inspected are included in the change-in-risk estimates in Table 3.4-1 and 3.4-2 in your submittal, how are the CCDP [conditional core damage probability], CLERP [conditional large early release probability], and the failure frequency estimated for low-safety-significant (LSS) welds?*

I&M Response to Request 4.b

For CCDP and CLERP, values of 1E-04 and 1E-05 were conservatively used. The rationale for using these values is that the change-in-risk evaluation process of N-716 is similar to that of the EPRI RI-ISI methodology. As such, the goal is to determine CCDP and CLERP threshold values. For example, the threshold values between High and Medium Consequence Categories is 1E-04 (CCDP) and 1E-05 (CLERP) and between Medium and Low Consequence Categories are 1E-06 (CCDP) and 1E-07 (CLERP) from the EPRI RI-ISI Risk Matrix. Using these threshold values streamlines the change-in-risk evaluation as well as stabilizes the update process. For example, if a CCDP changes from 1E-05 to 3E-05 due to an update, it will still be below the 1E-04 threshold value and the change-in-risk evaluation would not need to be updated.

The above values were compared to the CNP internal flooding study. The CCDPs for in-scope LSS Class 2 piping previously being inspected is less than 1E-04 and there were no containment bypass breaks; therefore, a 0.1 conditional large early release frequency is reasonable. The values are consistent with and conservatively above any CCDP value obtained for CNP in-scope Class 2 piping, and the CLERP value is appropriately scaled.

With respect to assigning failure potential for LSS piping, the criteria are defined by Table 3 of N-716. That is, those locations identified as susceptible to FAC (or another mechanism and also susceptible to water hammer) are assigned a high failure potential. Those locations susceptible to thermal fatigue, erosion-cavitation, corrosion, or stress corrosion cracking are assigned a medium failure potential and those locations that are identified as not susceptible to degradation are assigned a low failure potential.

In order to streamline the N-716 application, a review was conducted to verify that the LSS piping was not susceptible to FAC or water hammer. This review was conducted similar to that done for a traditional RI-ISI application. Thus, the high failure potential category is not applicable to LSS piping. In lieu of conducting a formal degradation mechanism evaluation for all LSS piping (e.g., to determine if thermal fatigue is applicable), these locations were conservatively assigned to the medium failure potential (denoted as "Assume Medium" in Reference 1, Table 3.4-1) for use in the change-in-risk assessment.

NRC Request 5

Under Section 3.3 on page 8, your submittal identifies 4 primary guidelines on selecting inspection locations, or 6 guidelines if each sub-bullet in (1) is counted as a guideline. Please describe briefly how each of these six guidelines was applied (e.g., how many inspections were influenced by the guideline and if application of the guideline resulted in changes to the original locations) when you were selecting inspection locations. Also, were any inspections added due to change-in-risk considerations?

I&M Response to Request 5

The process of defining the inspection population of an N-716 application is an iterative process. The first step is to define the scope of HSS welds on a per system basis. As a starting point, N-716 requires that 10% of the HSS welds, on a per system basis, be selected for inspection (see attached Table 5-1, column entitled "HSS"). The next step is to assure that 10% of Class 1 welds are selected (see attached Table 5-1, column entitled "Class 1"). It should be noted that a lesson learned from the Reference 1 proposed alternative is that this requirement could be more clearly stated in N-716. It is I&M's understanding that N-716 will be revised to reflect this and other lessons learned, as applicable. The next step is to assure that 25% of locations identified as potentially susceptible to some type of degradation mechanism be selected (see attached Table 5-1, column entitled "DMs"). The next step is to confirm that two thirds of the identified inspections for the RCPB are within the first isolation valve or move inspections from between the two isolation valves to within the first isolation valve to compensate, if necessary (see attached Table 5-1, column entitled "$RCPB^{IFIV}$"). The next step is to confirm, or select if necessary, that 10% of the RCPB that lies outside containment is inspected (see attached Table 5-1, column entitled "$RCPB^{OC}$"). Finally, inspections are chosen so that 10% of the break exclusion region (BER) populations are chosen (see attached Table 5-1, column entitled "BER"). Again, this may have already been accomplished by the preceding criteria, but needs to be confirmed or adjusted accordingly.

Depending upon how the element selection process is ordered, it may be necessary to iterate once or twice to assure the criteria are met. Because of rounding up, the selection being done on a system-by-systems basis, and the multiple criteria, it is expected that a greater than 10% inspection population will be attained (e.g., CNP examined 10.2% for Unit 1 and 10.1% for Unit 2).

With respect to change-in-risk considerations, no changes to the number or locations of inspections were required.

Table 5-1 (1)

Scope Selection and Weld Count

System	Unit	Selections	HSS	Class 1	DMs	$RCPB^{IFIV}$	$RCPB^{OC}$	BER
Reactor Coolant	Unit 1	Required	67 of 662	67 of 662	8 of 30	45	n/a	n/a
		Actual	67 of 662	67 of 662	16 of 30	67	n/a	n/a
	Unit 2	Required	67 of 669	67 of 669	7 of 26	45	n/a	n/a
		Actual	67 of 669	67 of 669	12 of 26	67	n/a	n/a
Containment Spray	Unit 1	Required	7 of 70	7 of 70	8 of 32 (2)	5	n/a	n/a
		Actual	7 of 70	7 of 70	7 of 32 (2)	7	n/a	n/a
	Unit 2	Required	7 of 64	7 of 64	9 of 34 (2)	5	n/a	n/a
		Actual	7 of 64	7 of 64	7 of 34 (2)	7	n/a	n/a
Residual Heat Removal	Unit 1	Required	5 of 48	3 of 22	n/a	2	n/a	n/a
		Actual	5 of 48	5 of 22	n/a	2	n/a	n/a
	Unit 2	Required	6 of 55	3 of 27	n/a	2	n/a	n/a
		Actual	6 of 55	6 of 27	n/a	2	n/a	n/a
Safety Injection	Unit 1	Required	45 of 442	45 of 442	12 of 47	8 of 32 (3)	1 of 9	n/a
		Actual	45 of 442	45 of 442	14 of 47	8 of 32 (3)	2 of 9	n/a
	Unit 2	Required	46 of 457	46 of 457	11 of 43	8 of 32 (3)	1 of 9	n/a
		Actual	46 of 457	46 of 457	12 of 43	8 of 32 (3)	2 of 9	n/a
Feedwater	Unit 1	Required	22 of 214	n/a	2 of 8	n/a	n/a	n/a
		Actual	22 of 214	n/a	2 of 8	n/a	n/a	n/a
	Unit 2	Required	20 of 200	n/a	2 of 8	n/a	n/a	n/a
		Actual	20 of 200	n/a	2 of 8	n/a	n/a	n/a

(1) For columns entitled "HSS," "Class 1," "DMs," "$RCPB^{OC}$," and "BER," the information provided is in the format of number of inspections per population of welds (e.g., a 10% requirement for a population of forty (40) welds would be "4 of 40"). For the column entitled $RCPB^{IFIV}$, this criterion is that 2/3 of the Class 1 inspections be inside the first isolation valve. Thus, this column identifies, on a "per system" basis, how many inspections were required per this criterion (row entitled "Required") and how many were actually selected to meet this criterion (row entitled "Actual").

(2) Per Section 4(b)(1) of N-716, a minimum of 25% of the population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected for examination. Per Section 4(b)(2), if the examinations selected per Section 4(b)(1) exceed 10% of the total number of high safety significant welds, the examinations may be reduced by prorating among each degradation mechanism and degradation mechanism combination, to the extent practical, such that as least 10% of the high safety significant population is inspected. This requirement was applied to the containment spray system.

(3) A modified element selection approach was implemented for the safety injection system based on lessons learned to address the requirement that 2/3 of the Class 1 examinations be located between the first isolation valve (i.e., isolation valve closest to the reactor pressure vessel (RPV)) and the RPV per Section 4(c) of N-716. For CNP, only 32 of 442 Class 1 welds for Unit 1 and 32 of 457 Class 1 welds for Unit 2 are located inside the first isolation valve. A 25% sampling of the total number of welds located inside the first isolation valve was alternatively selected for examination.

<u>NRC Request 6</u>

The relationship between N-716's guideline that "any piping segment whose contribution to core damage frequency (CDF) is greater than 1E-6/year is a high safety significant (HSS) segment," and the EPRI topical guidelines for safety significant categorization is unclear. For example, a low consequence segment in the EPRI Topical methodology has a CCDP less than 1E-6, an identical numerical value but a different metric than the 1E-6/year guideline in N-716. Page 3-8 in the EPRI Topical provides an explanation that the CCDP and conditional large early release probability (CLERP) ranges were selected "to guarantee that all pipe locations ranked in the low consequence category do not have a potential CDF impact higher than 1E-8 per year or a potential large early release frequency (LERF) impact higher than 1E-9 per year." Inspection of Table 3.1-1 and 3.1-2 in your submittal also indicates that there are no entries in the "CDF > 1E-6" column indicating that no segments in the CNP Units 1 and 2 flooding PRA exceeded this guideline.

a. *The N-716 code case Section 2(a)(5) does not include a LERF guideline analogous to the CDF guideline, and Table 3-1 in your submittal includes a column for CDF but not for LERF. Please explain why a LERF guideline is not included as a guideline in parallel with CDF.*

<u>I&M Response to Request 6.a</u>

I&M agrees that most PRA applications with a CDF guideline include a LERF guideline. Therefore, I&M proposes to add a LERF guideline of 1E-07/year to the requirements of Section 2(a)(5) of N-716. Additionally, I&M has reviewed LSS piping (e.g., non HSS Class 2, Class 3, and non-nuclear safety (NNS) piping) against the new LERF requirement. As a result of this review, I&M has confirmed that, in addition to having a CDF contribution of less than 1E-06/year, this piping also has a LERF contribution of less than 1E-07/year.

b. *Please provide a discussion justifying the guideline value for CDF selected in Section 2(a)(5) in N-716 (i.e., 1E-6/year).*

<u>I&M Response to Request 6.b</u>

As discussed in the response to NRC Request 6.a, I&M has added a criterion for LERF of 1E-07/year.

From a practical perspective, the criterion used in Section 2(a)(5) of N-716 has two potential impacts. Each is discussed below.

1. Class 2 Piping

   Any piping that has inspections added or removed per this code case, regardless of the value of this criterion, is required to be assessed as to its impact on risk. This risk impact analysis is conducted on an individual system basis, which includes the cumulative effect of LSS Class 2 piping currently being inspected. The change-in-risk acceptance criteria on a system basis are defined as 1E-07/year (CDF) and 1E-08/year (LERF). These criteria are derived from RG 1.174 and were approved by the NRC in EPRI TR-112657. If the change-in-risk acceptance criteria are not met, additional inspections are to be defined until these criteria are met (N-716 Section 5(d)). Therefore, regardless of the number of segments (or inspections) that fall below these criteria, unacceptable risk changes will not occur and the safety objectives of risk-informed regulation will be met.

   The change-in-risk analysis could be conducted without the benefit of these criteria (i.e., Section 2(a)(5) of N-716 and LERF per 6.a) and shown to have acceptable changes in plant risk. In fact, this was demonstrated in Reference 4 where eight plants (four boiling water reactors and four pressurized water reactors (PWRs)) were compared to the N-716 criteria. N-716 was shown to provide for more inspections than traditional RI-ISI approaches even when the criterion of Section 2(a)(5) was not used. As expected, the change-in-risk acceptance criteria of 1E-07/year (CDF) and 1E-08/year (LERF) were met for these eight plants. However, implementation of this ancillary criterion (Section 2(a)(5) of N-716 and LERF per NRC Request 6.a) provides increased confidence that the change-in-risk acceptance criteria will be met without the need for additional inspections as would be required by Section 5(d) of N-716. Thus, any risk outliers, if they exist in Class 2 piping (e.g., piping that exceeds the Section 2(a)(5) criterion and LERF per NRC Request 6.a), would require that, on a plant-specific basis, piping be added to the scope of HSS piping and subjected to inspection.

2. Class 3 / NNS Piping

   Currently, there are no ASME Code Section XI NDE requirements for this piping. As such, use of this ancillary criterion (Section 2(a)(5) of N-716 and LERF per NRC Request 6.a), regardless of its value, can only result in a reduction in plant risk further supporting the safety objectives of risk-informed regulation. These additional inspections would be imposed on piping identified by the criterion of Section 2(a)(5) of N-716 and LERF per NRC Request 6.a, and cannot be used to reduce inspections in other HSS piping (see N-716, Section 4(b)).

From a more global perspective, the ancillary criteria of Section 2(a)(5) of N-716 and of LERF per NRC Request 6.a provide additional criteria that can only potentially increase the scope of HSS locations (i.e., will only increase the number of inspections). Although the criteria of Sections 2(a)(1) through 2(a)(4) of N-716 were created based on the large number of risk-informed applications performed to date, Section 2(a)(5) of N-716 and LERF per NRC Request 6.a were added as a defense-in-depth measure to N-716 to provide a method of ensuring that any plant-specific locations that are important to safety are identified.

Adopting RI-ISI programs permits a reduction in inspection by focusing inspections on the more important locations. Use of this ancillary guideline and a technically adequate, plant-specific, flooding evaluation to identify relatively important locations (e.g., Class 2, 3, or NNS piping) provides additional confidence that inspections will be focused on the more important locations.

According to the guidelines in RG 1.174, plant changes (permitting the reallocation of resources) that increase risk less than 1E-06/year (CDF) and 1E-07/year (LERF) would normally be considered very small and acceptable as long as the other principles are satisfied. This is considered to be a reasonable metric for identifying significant pipe segments since the potential reduction in CDF and LERF from inclusion of such segments in the ISI program would also be very small. Additionally, use of the guideline value of 1E-06/year for CDF (1E-07/year for LERF) taken together with the system level change-in-risk limits of 1E-07/year for CDF (1E-08/year for LERF) provides additional assurance that plant-specific application of N-716 will meet the acceptance criteria of Region III in Figures 3 and 4 of RG 1.174, assuring any increase would be small and consistent with the intent of the NRC's Safety Goal Policy Statement (Reference 5).

Finally, traditional RI-ISI approaches can be applied on a partial scope basis. That is, many plants have applied RI-ISI to Class 1 piping only. Thus, these plants have not witnessed the additional safety benefit of identifying and inspecting Class 2, 3, or NNS piping per criterion Section 2(a)(5) of N-716 and LERF per the response to NRC Request 6.a.

c. *Please provide a list of all the piping segments that were compared to the >1E-6/year criteria along with the CDF and LERF estimates, the pipe failure frequency, and the CCDP and conditional large early release probability for each segment.*

I&M Response to Request 6.c

The scope of piping reviewed against this criterion consisted of Class 2 piping not classified as HSS (e.g., BER), Class 3, and NNS piping. The BER piping at CNP is limited to NPS less than (<) 4-inch portions of the main steam blowdown system and the chemical and volume control system, and is excluded from the N-716 evaluation. There will be no changes to the current BER examination schedule. The updated CNP flooding PRA was used to conduct this comparison. The updated CNP flooding PRA was performed consistent with the Reference 6 guideline. That is, the internal flooding PRA was performed by defining flood zones, identification of flood zone contents (e.g., important equipment), flood zone flood sources and propagation pathways, a qualitative screening analysis, and a quantitative analysis of the remaining potentially important flooding scenarios.

The bounding, screening, and quantitative analyses resulted in all flood zones and groups falling below the 1E-06 CDF criterion except two dominant contributors. The first involved a failure of a fire protection line in the Auxiliary Building which was postulated to flood the electrical switchgear Train A direct current (DC) distribution panel room (CDF contribution of 6.11E-06). The second involved failures of the circulating water system in the condenser pit (CDF contribution of 3.75E-06).

Based on the above, more detailed analysis was conducted that reflected a plant modification (fire protection line) and more realistic analyses (e.g., revised Human Error Probability (HEP)) so that these scenarios now fall below the 1E-06 CDF criterion.

With respect to LERF, see the response to NRC Request 6.a.

d. *Please provide any observations made during any independent reviews of the CNP flooding PRA or observations from the internal events review that are also applicable to the flooding analysis.*

Please describe how these observations have been resolved such that there is confidence that segments that have a CDF greater than the guideline value have been identified.

I&M Response to Request 6.d

There was only one internal flooding related "Fact and Observation" (F&O) from the CNP PRA peer review process. That F&O was as follows:

"Flood barriers were not treated probabilistically. All flood barriers were assumed to function. Back flow through drains was also not assumed to occur."

The flooding analysis screened away all rooms except the turbine building basement. The screening criteria considered pipe spray mode only (i.e., no ruptures), which resulted in the screening out of all rooms.

This is Level A significance, since the flooding CDF is very low (2E-07), based on screening away of all rooms using erroneous criteria.

That single F&O was resolved by generation of an updated CNP flooding PRA, which, as noted in the response to 6.c, was performed consistent with Draft ASME RA-Sa-2003, Addenda B, flooding study guidelines. That is, the internal flooding study was performed by defining flood zones, identification of flood zone contents (e.g., important equipment), flood zone flood sources and propagation pathways, a qualitative screening analysis, and a quantitative analysis of the remaining potentially important flood scenarios.

e. *Page 3 of your submittal states that internal flooding was recently addressed (2006) to complete the effort to address all Westinghouse Owners Group certification Level A and B F&Os. To the extent not discussed in the response to RAI 6.d, please explain what "addressed" means. Were changes made to a flooding analysis? If changes were not made, how are the F&Os addressed? How were the changes that were made, or the explanation for not requiring changes, reviewed for technical adequacy?*

I&M Response to Request 6.e

"Addressed," as used in the submittal, means that the CNP internal flooding analysis was updated to meet the intent of the flooding F&O, as noted in the response to NRC Request 6.d. The prior CNP flooding analysis was the analysis from the original CNP individual plant examination (IPE) flooding evaluation. The flooding update was accomplished over the 2005-2006 time frame, using the Draft ASME RA-Sa-2003, Addenda B, technique and criteria for determining flooding susceptibility, screening areas/systems from consideration. This update was captured in accordance with plant procedures for non-safety related calculations. These calculations, while not safety related, underwent an in-house, independent review (e.g., calculation preparer and

reviewer) process. The update resulted in substantial changes in the flooding evaluation from the initial IPE evaluation upon which the flooding F&O was based. The updated flooding evaluation relied on the 2005 internal events PRA model to address various flooding scenarios that were not screened out of consideration.

This updated PRA flooding analysis initially found two scenarios that produced a CDF in excess of 1E-06. One scenario involves a fire hose station mounted in a small room housing the Train A DC distribution panels, which had the potential to impact the Train B DC distribution system. A minor plant hardware modification to seal the associated Train B DC panel subsequently reduced the impact of this scenario, removing it from the list of scenarios requiring further detailed analysis. The second involved failures of the circulating water system in the condenser pit (CDF contribution of 3.75E-06). Revision of an HEP reduced the CDF from this scenario (see response to NRC Request 6.c).

NRC Request 7

Page 12 describes how the CCDP and CLERP of different types of HSS pipe breaks are estimated in support of the change-in-risk estimates. Some values appear to be derived from representative sequences from the PRA models while others are directly estimated. For example, bounding values for pipe breaks that result in isolable LOCAs [loss of coolant accidents] are directly estimated as the product of the CCDP from unisolable LOCAs and the probability of a motor operated valve failing to close on demand. Direct estimation can be very analyst-specific and essentially bypasses the PRA peer review process upon which the NRC relies to minimize the staff review of the plant-specific PRA for each risk-informed submittal.

a. *Please identify events modeled in the CNP PRA that are similar to the directly estimated values on page 12 of your submittal or further clarify why the PRA cannot be used to develop the required estimates (these appear to be ILOCA, PLOCA, PILOCA-OC, and PILOCA-IC). If applicable events in the PRA can be identified, please provide a description of these events and the bounding CCDP and CLERP values for these types of breaks derived from the PRA.*

I&M Response to Request 7.a

The CNP PRA does not explicitly model potential and isolable LOCA events because the LOCA initiators in the PRA do not distinguish break location. The N-716 methodology must evaluate these segments individually; thus, it is necessary to estimate their contribution by taking the LOCA CCDP and multiplying this by the valve failure probability.

b. *In the Table on page 12, please describe the difference between row 2, isolable LOCA (assumed to be inside containment), and row 5, potentially isolable LOCA inside containment. In what category would a pipe break that relied on a MOV [motor operated valve] that does not close automatically but that could be closed remotely by a manual action be placed?*

<u>I&M Response to Request 7.b</u>

The isolable LOCA (row 2) is a segment downstream of an air operated valve (AOV) that automatically isolates on low pressurizer level (template table on Page 12 has typographical error indicating MOV rather than AOV). MOVs have a failure probability that is slightly larger than AOVs, which therefore provides a slightly higher CCDP. For conservatism, the high consequence rank is maintained. The potential LOCA (row 5) is a segment downstream of a normally closed valve, in this case a check valve. Thus, the CCDP is estimated as the product of check valve rupture and LOCA CCDP. This particular segment also includes some piping downstream of an MOV that does not get an automatic signal, thus credit for another isolation valve was not taken. Since there is uncertainty with regard to the operators ability to detect this break location in time to prevent a LOCA, operator action was not credited.

c. *Row 6, "Class 2 SDC - IC" states that the CCDP and CLERP are "[e]stimated based on a loss of shutdown cooling during mid-loop operation." Are these values intended to develop the safety significance of these segments during shutdown, or as surrogates for power operation. If these values are intended as surrogates for power operation, please explain why these values are reasonable surrogates. If not intended as surrogates for power operation, how was the safety significance of these segments during power operation addressed.*

<u>I&M Response to Request 7.c</u>

Since the shutdown cooling piping inside containment has two normally closed valves during power operation, the CCDP for power operation is clearly <1E-04 as summarized below (this result is consistent with a number of RI-ISI applications):

> The potential LOCA scenarios require two valves in series to fail open which would be multiplied by LOCA CCDP.
>
> The injection paths could also fail during an accident demand, but there are redundant backup injection paths and the CCDP for this event required the probability of challenge times the CCDP for the backup paths.

As a result, it was assumed that pipe break during shutdown operation could be more important and it was assumed to have a 1E-04 CCDP based on qualitative reviews on several previous RI-ISI applications. The reference to mid-loop could be deleted as it could be misleading in that the table was meant to provide a general reference to shutdown configurations, not just mid-loop.

d. *The last row in the Table on page 12 includes an entry labeled "Class 2 LSS". What characteristics result in a "Class 2 LSS" designation? The same entry further states that the CCDPs and CLERPs of pipe ruptures associated with these welds are "[e]stimated based on upper bound for Medium Consequence." Please provide a discussion explaining why selecting these values is appropriate.*

<u>I&M Response to Request 7.d</u>

The "Class 2 LSS" designation is used to identify those Code Class 2 locations that are not HSS because they do not meet any of the five HSS criterions of section 2(a) of N-716 (e.g., not part of the BER scope). With respect to CCDPs/CLERPs, see response to NRC Request 4.b.

e. *The ASME standard RA-Sa-2003, element IE-C12 discusses the evaluation of the likelihood of a interfacing system LOCA. In which category does the interfacing system LOCA belong in your Table on page 12?*

<u>I&M Response to Request 7.e</u>

The Potentially Isolable LOCA Outside of Containment (PILOCA-OC) break location on the Page 12 table applies to the category of breaks in piping connected to RCPB outside containment. For CNP, a 1.0 CCDP and a 1.0 CLERP were used for piping outside containment and connected to the RCPB (the CCDP only credited valve failures required to cause the LOCA outside containment). A conservative estimate of CCDP can be used for this application as long as it supports the determination that the change-in-risk is low. More realistic calculations would only be required if these simplified approaches indicated potentially unacceptable risk increases.

<u>NRC Request 8</u>

Under Section 2.2 on Page 4, you state that, "[t]he requirements of MRP-139 will be used for inspection and management of primary water stress corrosion cracking (PWSCC) susceptible welds and will supplement the RIS_B Program selection process." Please describe what is meant by "supplement." How will the PWSCC degradation mechanism be addressed, as any other mechanism or differently? How will any inspections that might be required by MRP-139 be credited in the RIS_B program?

I&M Response to Request 8

The MRP-139 inspection schedule will be followed. All of the pressurizer nozzle butt welds have had weld overlays installed. The inspection schedule will be in accordance with Relief Requests ISIR-15, 20, and 21 (References 7, 8, and 9). This requires 25% of the overlays to be inspected during the interval. The remaining butt welds are in the Unit 1 reactor vessel nozzles. The reactor vessel nozzles will be inspected and/or mitigated using the guidance in MRP-139.

NRC Request 9

Is the guideline to examine a minimum 10% of all HSS welds, or 10% of all HSS butt welds, or 10% of all HSS butt welds >= 4 NPS?

I&M Response to Request 9

See response to NRC Request 3.a.

NRC Request 10

Under Section 3.4 on Page 11, your submittal states "the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements." Consistent with this, the total change-in-risk in the two tables on page 14 is always negative. Is Cook committing to ensuring that these total risk numbers will be maintained at or below 0 as it monitors the program over time as described in Section 4 of your submittal?

I&M Response to Request 10

The change-in-risk will meet the acceptance criteria per section 5(d) of Code Case N-716. This is consistent with the acceptance criteria in EPRI TR-112657.

NRC April 9, 2007, RAI (Reference 3)

NRC Request 1

Footnote 2 for the table on page 9 of the licensee's submittal indicates that 240 Class 2 welds are HSS, yet only 22 welds are selected for inspection at Unit 1 and 228 Class 2 welds are HSS yet only 20 welds are selected for inspection at Unit 2. These selections do not appear to meet the 10% requirement for HSS locations. Please explain this discrepancy.

I&M Response to Request 1

Per Section 4 of N-716, 10% of the high safety significant welds shall be selected for examination. Subparagraphs 4(a) through 4(f) of N-716 specify how the 10% sampling shall be distributed. These requirements are addressed in the response to NRC Request 5 from the March 29, 2007, RAI. N-716 does not require that a 10% sampling of the ASME Code Section XI Class 2, welds designated as HSS be selected for examination. As stated in the response to NRC Request 5 from the March 29, 2007, RAI, it is I&M's understanding that N-716 will be revised to explicitly state that a 10% selection of Class 1 welds is required, but this same requirement does not apply to Class 2. This selection philosophy, as it pertains to Class 1 and 2 piping welds, is identical to that implemented in EPRI TR-112657, which has been approved by the NRC.

NRC Request 2

Section 5 of the licensee's submittal states that the licensee will implement the RIS_B program during the plant's third period of the current (third) inspection interval by performing 66% of the inspection locations selected for examination per the RIS_B process for each unit. Describe how the licensee will determine which examinations to perform during the remainder of the third 10-year ISI interval.

I&M Response to Request 2

Prior to developing the RIS_B Program, CNP had planned to inspect locations scheduled for examination in the traditional ASME Code Section XI inspection program. Examination activities during refueling outages are planned far in advance. In general, only designated plant areas and components are accessible for examination during a given refueling outage due to other ongoing plant maintenance and modification activities. Hence, any location previously scheduled for examination in the third period via the traditional program will remain scheduled for examination in the third period if the location has also been selected for RIS_B Program purposes. To complete the sample size, additional locations will be selected, if necessary, to achieve equal representation of

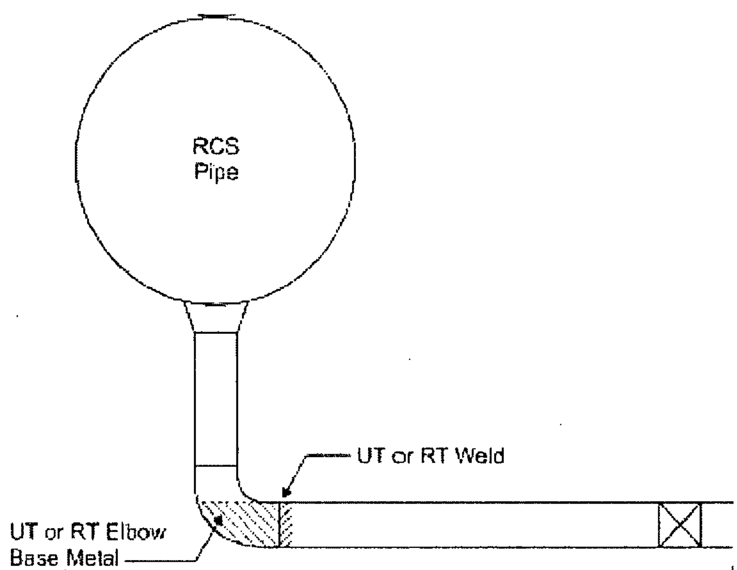
the degradation mechanisms. Other factors such as accessibility and scaffolding requirements will also be factored into the selection process.

NRC Request 3

Please describe how volumetric examinations will be performed. At a minimum, will volumetric examinations include the volume required for ASME Section XI examinations? Will ASME Section XI, Appendix VIII qualified examiners and procedures be used for all volumetric exams? Will the examination volume be scanned for both axial and transverse indications for all exams? Please describe and justify your answers.

I&M Response to Request 3

Volumetric examinations will be performed as required by Table 1 of N-716. The table requires an examination volume as defined in the ASME Code Section XI IWB figures. This would require examination of at least the ASME Code Section XI volume (more volume may be required based on the notes on Table 1 of N-716). N-716 does not take any exceptions to the paragraphs of ASME Code Section XI that govern volumetric examinations and I&M's proposed alternative request does not take exception to any 10 CFR limitations. Therefore, I&M will examine these welds using the same personnel and procedure requirements as a traditional ASME Code Section XI piping volumetric examination.



Typical Inspection Regions

NRC Request 4

Please describe how preservice examinations will be performed for repair/replacement activities. Include what repair/replacement items will receive preservice examination.

I&M Response to Request 4

For preservice examinations, I&M will follow the rules contained in Section 3.0 of N-716. Welds classified HSS require preservice inspection. The examination volumes, techniques, and procedures shall be in accordance with Table 1 of N-716. Welds classified as LSS do not require preservice inspection.

NRC Request 5

Page 10 discusses additional examinations. Please describe what will be used to perform the engineering evaluation to determine the cause of any unacceptable flaw or relevant condition. Recent industry practice has been to perform corrective actions (i.e., overlays, replacement, etc.) prior to a root cause being determined (e.g., use of a qualified procedure and personnel).

I&M Response to Request 5

Any unacceptable flaw will be evaluated per the requirements of ASME Code Section XI, IWB-3500, and/or IWB-3600. As part of performing an evaluation to IWB-3600, the degradation mechanism that is responsible for the flaw will be determined and accounted for in the evaluation. If the flaw is found unacceptable for continued operation, it will be repaired in accordance with IWB-4000 and/or applicable ASME Code Section XI Code Cases. The need for extensive root cause analysis beyond that required for IWB-3600 evaluation will be dependent on practical considerations, such as the practicality of performing additional NDE or removal of the flaw for further evaluation during the outage.

a. *In some cases no materials are removed for metallurgical analysis. Please discuss the process used for this engineering evaluation, how will it be documented, and will the Nuclear Regulatory Commission be involved in the process?*

I&M Response to Request 5.a

The process for ordinary flaws is to perform the evaluation using ASME Code Section XI. If the flaw meets the criteria, then it is noted and the appropriate successive examinations are scheduled.

The NRC is involved in the process at several points. For preemptive overlays, a relief request is usually needed for the design and installation. Should the flaw be discovered during the examination, a notification per 10 CFR 50.72 or 10 CFR 50.73 may be made. IWB-3600 requires the evaluation to be submitted to the NRC. Finally, NIS-1 and NIS-2 forms summarizing the inspections and repairs performed during the outage are submitted to the NRC.

b. *Discuss what process will be used to perform fracture mechanics evaluations.*

I&M Response to Request 5.b

ASME Code Section XI, IWB-3600, provides the rules for flaw evaluation and fracture mechanics. The results of the evaluation are required to be submitted to the NRC.

c. *Discuss under what conditions would there be no additional examinations. Discuss how the licensee will document its justification.*

I&M Response to Request 5.c

If the flaw is original construction or otherwise acceptable, ASME Code Section XI rules do not require any additional inspections. If the nature and type of the flaw is service induced, then similar systems or trains will be examined. The documentation requirements will be documented in the Corrective Action Program and a summary will be submitted in the NIS-1 package.

NRC Request 6

Page 10, Section 3.3.2 "Program Relief Requests," provides guidance. For program relief requests the licensee refers to the process outlined in Reference 2. Recently there have been problems associated with giving relief [for] limited examinations from risk-informed ISI program items. For limited examinations of RIS_B selected items please describe the process for assessing limited examination coverage. Discuss whether additional examinations will be performed, and whether additional techniques will be used to improve examination coverage. Discuss how the effect on risk of the incomplete examination coverage will be assessed. In what time frame will relief requests be submitted?

I&M Response to Request 6

I&M will calculate coverage and use additional examinations or techniques in the same manner it has for traditional ASME Code Section XI examinations. Experience has shown this process to be weld-specific (e.g., joint configuration). As such, the effect on risk, if any,

will not be known until that time. Relief requests will be submitted when the condition is identified.

NRC Request 7

Page 10 also discusses that Relief Requests ISIR-005 and ISIR-006 will be withdrawn. Please discuss why these requests will be withdrawn. Also the licensee states that pipe-to-flue head welds in the feedwater system are included in the scope that is designated high safety significant, yet have not been selected for examination. Describe why none of these welds are selected for examination.

I&M Response to Request 7

During the development of the risk-informed template process, the NRC requested that licensees address the impact of the risk-informed application on existing plant ISI Program relief requests. The NRC requested notification in the template submittal of any relief requests that would be modified or withdrawn as a result of the change in inspection philosophy. For the CNP N-716 application, this impact is addressed in Section 3.3.2 of the plant template submittal. Further explanation is provided below.

Feedwater Pipe to Flued Head Welds (ISIR-005) – These locations are included in the system boundaries (i.e., steam generator to the outer containment isolation valve) designated HSS, but were not selected for RIS_B examination. I&M did not choose these locations as part of the 10% HSS examination sampling required for the feedwater system because they are inaccessible and because no degradation mechanisms were identified. In addition, it should be noted that these locations are not mandatory selections per the 1989 Edition of ASME Code Section XI, the CNP code of record for the current third ten-year interval ISI program. As such, a relief request is not required.

Main Steam Pipe to Flued Head Welds (ISIR-006) – The main steam system in its entirety is designated LSS and is, therefore, not subject to RIS_B examination. Similar to the above, it should be noted that these locations are not mandatory selections per the 1989 Edition of ASME Code Section XI, the CNP code of record for the current third ten-year interval ISI program. As such, a relief request is not required.

NRC Request 8

Section 3.3.2 states that an attempt was made to select locations for examination such that a minimum >90% coverage is attained. Discuss how this attempt was conducted. If less than 90% examination is completed, discuss whether additional weld(s) will be examined to compensate for the limited examination coverage.

I&M Response to Request 8

As discussed in EPRI TR-112657, accessibility is an important consideration in the element selection process of an RI-ISI application. As such, for the CNP N-716 application, locations have generally been selected for examination where the desired coverage is achievable. This is typically accomplished by utilizing previous inspection history, plant access considerations, and knowledgeable plant personnel. However, some limitations will not be known until the examination is performed since some locations will be examined for the first time.

In addition, other considerations may take precedence and dictate the selection of locations where greater than 90% examination coverage is physically impossible. This is especially true for element selections where a degradation mechanism may be operative (e.g., risk categories 1, 2, 3, and 5 of EPRI TR-112657). For these locations, elements are generally selected for examination on the basis of predicted degradation severity. For example, in the emergency core cooling system injection lines of PWRs, the piping section immediately upstream of the first isolation check valve is considered susceptible to intergranular stress corrosion cracking, assuming a sufficiently high temperature and oxygenated water supply. The piping element (pipe-to-valve weld) located nearest the heat source will be subjected to the highest temperature (conduction heating). As such, this location will generally be selected for examination since it is considered more susceptible than locations further removed from the heat source, even though a pipe-to-valve weld is inherently more difficult to examine and obtain full coverage than most other configurations (e.g., pipe-to-elbow weld). In this example, less than 90% coverage of this location will yield far more valuable information than 100% coverage of a less susceptible location.

For locations with no identified degradation mechanisms (i.e., similar to risk category 4 of EPRI TR-112657), a greater degree of flexibility exists in choosing inspection locations. As such, if at the time of examination an N-716 element selection is found to be obstructed, a more suitable location may be substituted.

Therefore, I&M will review each instance of limited coverage and take the appropriate steps (e.g., relief requests) consistent with its impact on the basis of the N-716 application.

References

1. Letter from J. N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Request for Approval of Risk-Informed Inservice Inspection Program for Class 1 and 2 Piping American Society of Mechanical Engineers Code, Category B-F, B-J, C-F-1, and C-F-2 Piping Welds," AEP:NRC:6055-09, Accession Number ML062850540, dated September 29, 2006.

2. Electronic Communication from P. S. Tam, NRC, to M. K. Scarpello, I&M, "Draft RAI on D. C. Cook Risk-Informed ISI Program (TAC Nos. MD3137, 8)," Accession Number ML070890463, dated March 29, 2007.

3. Electronic Communication from P. S. Tam, NRC, to M. K. Scarpello, I&M, "D. C. Cook – Draft RAI Questions re: Risk-Informed ISI Program (TAC Nos. MD3137, 8)," Accession Number ML070990628, dated April 9, 2007.

4. "Whitepaper in Support of Code Case N716," dated October 2005.

5. NRC, "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," *Federal Register*, Volume 51, Page 30028 (51 FR 30028), dated August 4, 1986.

6. ASME RA-Sa-2003, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Draft Addenda B, dated March 2005.

7. Letter from L. Raghavan, NRC, to M. K. Nazar, I&M, "Donald C. Cook Nuclear Plant, Unit 1 (DCCNP-1) – Alternatives Regarding Repair of Weld 1-PZR-23 on Pressurizer Nozzle to Valve Inlet Line (TAC No. MC6704)," Accession Number ML053220019, dated December 1, 2005.

8. Letter from L. Raghavan, NRC, to M. K. Nazar, I&M, "Donald C. Cook Nuclear Plant, Unit 2 (DCCNP-2) – Alternative Regarding Use of Preemptive Weld Overlays on Certain Dissimilar Metal Welds (TAC No. MC9305)," Accession Number ML070460121, dated March 1, 2007.

9. Letter from L. Raghavan, NRC, to M. K. Nazar, I&M, "Donald C. Cook Nuclear Plant, Unit 1 (DCCNP-1) – Alternative Regarding use of Preemptive Weld Overlays (PWOLs) on Certain Dissimilar Metal Welds (TAC No. MD2119)," Accession Number ML070720021, dated April 26, 2007.