VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 13, 2010

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	10-050A
NL&OS/ETS	R0
Docket Nos.	50-338/339
License Nos.	NPF-4/7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
ASME SECTION XI INSERVICE INSPECTION PROGRAM
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
RELIEF REQUEST N1-I4-RI-001 AND N2-I4-RI-001
REQUEST FOR ALTERNATIVE - IMPLEMENTATION OF A RISK-INFORMED INSERVICE INSPECTION PROGRAM BASED ON ASME CODE CASE N-716

In a February 23, 2010 letter (Serial No. 10-050), Dominion requested authorization to implement a risk-informed inservice inspection (RI-ISI) program based on the American Society of Mechanical Engineers (ASME) Code Case N-716, as documented in the Requests for Alternative N1-I4-RI-001 and N2-I4-RI-001 for Units 1 and 2, respectively. N1-I4-RI-001 and N2-I4-RI-001 were submitted in a template format. In an August 12, 2010 e-mail from Dr. V. Sreenivas, the NRC requested additional information to complete the review of the RI-ISI program. The attachment to this letter provides the requested information.

Dominion plans to implement this alternative for the entire 4th ISI Interval for North Anna Units 1 and 2. North Anna Unit 1's 4th 10-Year Interval began May 1, 2009 and will end April 30, 2019. North Anna Unit 2's 4th 10-Year Interval begins December 14, 2010 and will end December 13, 2020. Therefore, Dominion continues to request review and approval of N1-I4-RI-001 and N2-I4-RI-001 by February, 2011 in order to plan and complete the first period examinations.

If you have any questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Respectfully,

J. Alan Price
Vice President – Nuclear Engineering

Attachment

1. Response to Request for Additional Information - Relief Requests N1-I4-RI-001 and N2-I4-RI-001 with 3 Enclosures

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Attachment

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

INSERVICE INSPECTION PLAN
RISK INFORMED FOURTH INTERVAL RI RELIEF REQUESTS
N1-I4-RI-001 AND N2-I4-RI-001

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Background

By letter dated February 23, 2010, Virginia Electric and Power Company (Dominion), submitted for staff review and approval ISI Program Relief Requests N1-I4-RI-001 and N2-I4-RI-001, which request approval to use alternative risk-informed inservice inspection (RI-ISI) selection and examination criteria for Category B-F, B-J, C-F-1 and C-F-2 pressure retaining piping welds for the North Anna Power Station (NAPS) Units 1 & 2. To complete their review, the NRC staff requested the following additional information.

Probabilistic Risk Assessment Licensing Branch

The NRC has not endorsed EPRI (Electric Power Research Institute) Topical Report 1018427. The questions listed below address the quality of North Anna probabilistic risk assessment (PRA) model.

Question 1

A self assessment performed on the North Anna PRA model in August 2007 identified PRA modeling and documentation supporting requirements (SRs) where the PRA model did not meet Capability Category (CC) II of the ASME (American Society of Mechanical Engineers) PRA standard. In December 2009, a model update was performed to meet Category II of the ASME PRA standard and Regulatory Guide 1.200 Rev 1. Please identify and disposition any remaining differences with CC II requirements (i.e., open items) that may affect this application.

Dominion Response

A self assessment was performed on the North Anna PRA model in June 2010 identifying remaining open items that do not meet Regulatory Guide 1.200 Rev 1 Capability Category (CC) II requirements. These open items are identified and evaluated for their potential to impact the risk assessment performed for the RI-ISI program at North Anna. Specifically, the unmet supporting requirements (SRs) are considered for their ability to impact the quantification of a large break LOCA, which was the bounding case that was used for the change in risk analysis, or the internal flooding analysis, which was used for scope determination. Based on the evaluation in the table below, none of the open items identified in the current North Anna PRA model affect the inputs or results of this application.

Gap	Description	Self Assessment	Impact on the RI-ISI Application
AS-A4	For key safety functions (e.g., power restoration) identify operator actions to achieve the defined success criteria.	SR remains as NOT MET until 1) an human event probability (HEP) is added to the station Blackout (SBO) nodes for restoring the ECCS functions; and 2) text in section 2.3.3.1 is revised to clarify the need for operator action to restart ECCS functions.	1) The importance of the SBO Diesel is low with respect to flooding events and a large break LOCA event in the North Anna PRA model. The Risk Achievement Worth (RAW) of the SBO for flooding events is 1.00, and the SBO RAW for a large break LOCA event is 1.01. Based on this low risk worth, adding an HEP to the SBO nodes for restoring the ECCS functions would not impact the results of the risk assessment performed to support this application. 2) Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
AS-B5a	Define and model plant configurations and alignments that reflect dependencies.	The NAPS models credit use of the opposite unit systems, e.g., charging system and diesel-generators, for accident mitigation. However, no documentation was identified that would show how opposite unit outages were considered. For example, during a refueling outage, a Train-A outage may make charging or component cooling (CC) cross-tie unavailable for a significant period of time. Such unavailability values could reach 5% overall. If unavailability during opposite-unit outages is included in the overall system unavailability, then that could be stated in the accident sequence (AS) documentation.	For all of the crosstie systems, either the unavailability during refueling outages is accounted for in the PRA unavailabilities or the system/trains do not have significant unavailability during outages. The only exception are the electrical buses where the unavailability during at power operation is not included in the PRA model. The estimated unavailability is 1-2 days, which is less than 1E-2 change in estimated unavailability. This small change in unavailability would not impact the flooding evaluation or the quantification of a large break LOCA.
DA-D2	When using expert judgment document the rationale behind the choice of parameter values.	Documentation needs to be enhanced for the several cases where expert opinion is used. The expert opinion is reasonable and should not change.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

Gap	Description	Self Assessment	Impact on the RI-ISI Application
QU-B1	Identify method-specific limitations and features that could impact the results and applications.	Although key assumptions are documented, these do not include limitations of the quantification method or features that impact results (aside from references to code limitations, guidance documents and procedures).	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-F5	Identify method-specific limitations and features that could impact the results and applications.	Although key assumptions are documented, these do not include limitations of the quantification method or features that impact results (aside from references to code limitations, guidance documents and procedures).	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SC-A6	Include a discussion of operator actions assumed as part of the success criteria development, and how those actions are consistent with plant procedures and practices.	Some of the success criteria discussion includes general operator actions, but the discussion does not include procedures and not all event tree sections contain the discussion.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A2	Use results of plant walkdowns and plant personnel interviews (system engineers and operators) as a source of information for modeling the as-built, as-operated plant.	The Dominion PRA staff has performed many system walkdowns during the development and maintenance of the models. In addition, Dominion PRA staff works closely with North Anna system engineers and operators on nearly a daily basis while supporting the various risk informed programs. However, no formal documentation exists at this time to allow closure of these supporting requirements (SRs). It is NOT anticipated that not meeting this requirement will have a significant impact on the model.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-B15	Identify SSCs that may be required to operate in conditions beyond their environmental qualifications.	Currently, the NAPS PRA model does not distinguish between PZR PORVs failing to reclose on water or steam relief. See EPRI TR-1011047 "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief."	Including a specific failure probability for a pressurizer PORV failing to reseat after passing water would not impact the internal flooding evaluation or large break LOCA quantification, so this open item does not impact the RI-ISI application.

Gap	Description	Self Assessment	Impact on the RI-ISI Application
SY-B8	Use results of plant walkdowns and plant personnel interviews (system engineers and operators) as a source of information for modeling the as-built, as-operated plant.	The Dominion PRA staff has performed many system walkdowns during the development and maintenance of the models. In addition, Dominion PRA staff works closely with North Anna system engineers and operators on nearly a daily basis while supporting the various risk informed programs. However, no formal documentation exists at this time to allow closure of these SRs. It is NOT anticipated that not meeting this requirement will have a significant impact on the model.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
HR-G4	Base the time available to complete actions on appropriate realistic generic thermal-hydraulic analyses, or simulation from similar plants	Time windows for successful completion of actions in some instances may need to be updated (for example, those that are based on estimates made for the IPE).	As part of the 2009 model update, new MAAP runs were performed for some of the key operator actions. This includes runs to support the estimation of HEP-1FRH:1-11 and HEP-1FRH1-15-ONE, which are the only HEPs that could significantly affect the internal flooding evaluation. The HEP for transferring to hotleg recirculation, HEP-1ES1:4, is the only HEP important for large break LOCA quantification. Updating the timing for additional HEPs to meet CC II of this SR would not impact the risk assessment performed for the RI-ISI application.
HR-G5	Base the required time to complete actions for significant HFEs on action time measurements in either walkthroughs or talk-throughs of the procedures or simulator observations.	No documentation currently exists and this SR will remain NOT MET. As a footnote the timings are not expected to change significantly as they are based on comparisons with similar actions at Surry.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A4	Use results of plant walkdowns and plant personnel interviews (system engineers and operators) as a source of information for modeling the as-built, as-operated plant.	The Dominion PRA staff has performed many system walkdowns during the development and maintenance of the models. In addition, Dominion PRA staff works closely with North Anna system engineers and operators on nearly a daily basis while	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

Gap	Description	Self Assessment	Impact on the RI-ISI Application
		supporting the various risk informed programs. However, no formal documentation exists at this time to allow closure of these SRs. It is NOT anticipated that not meeting this requirement will have a significant impact on the model.	
AS-A7	Delineate accident sequence (e.g., Loss of RCP seal cooling) for each initiating event (e.g., transients).	SR is NOT MET until: 1) inclusion of consequential loss of RCP seal cooling for transients, and 2) documentation enhancement of the U1-RCPSL nodes.	1) Inclusion of consequential RCP seal cooling for transients would not affect the application because only the large break LOCA and flooding events were quantified. Consequential loss of RCP cooling is considered for flooding events in the North Anna PRA model, and does not apply for a large break LOCA scenario. 2) Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-E1	Identify key sources of model uncertainty.	Each PRA element notebook (IE, AS, SC, SY, DA, HR, LE) has identified potential sources of model uncertainty. A characterization of those sources of uncertainty and evaluation of the generic sources of uncertainty has not yet been completed however.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-F4	DOCUMENT key assumptions and key sources of uncertainty, such as: possible optimistic or conservative success criteria, suitability of the reliability data, possible modeling uncertainties (modeling limitations due to the method selected), degree of completeness in the selection of initiating events, possible spatial dependencies, etc.	Although the different element notebooks (IE, AS, SC, SY, etc.) do include specific assumptions related to the development of that element, there is no discussion in the QU.1 (input) and QU.2 (results) notebooks of the sources of uncertainty in the NAPS model, nor of the assumptions associated with those uncertainties.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

Gap	Description	Self Assessment	Impact on the RI-ISI Application
LE-D4	PERFORM a realistic secondary side isolation capability analysis for the significant accident progression sequences caused by SG tube release. USE a conservative or a combination of conservative and realistic evaluation of secondary side isolation capability for non-significant accident progression sequences resulting in a large early release. JUSTIFY applicability to the plant being evaluated. Analyses may consider realistic comparison with similar isolation capability in similar containment designs.	Secondary side isolation is explicitly and realistically modeled in the Level 1 System Analysis notebooks for pre-core damage consideration. However, secondary side isolation during a SGTR should also consider the additional number of demands on the relief valves in the progression to core damage. It is possible that some sequences considered "isolated" in the Level 1 analysis could be unisolated in the Level 2 analysis. Also, version 4 of the MAAP code provides better SGTR analysis than had been used for the IPE with version 3 of the code.	Not Significant. This is with regards to SGTR initiating event and would not impact the flooding evaluation or the quantification of a large break LOCA.
QU-E2	IDENTIFY key assumptions made in the development of the PRA model.	The QU.1 (input) notebook indicates that key modeling assumptions are documented in Part II of the PRA model notebook, but this part has not yet been developed (although some key assumptions may be available in the IPE submittal). The different element notebooks (IE, AS, SC, SY, etc.) do include specific assumptions related to the development of that element, but there is typically no discussion of the sources of uncertainty those assumptions relate to and the impacts of those assumptions.	Not Significant. As part of the 2009 model update, within each PRA element notebook (IE, AS, SC, SY, DA, HR, LE), potential sources of model uncertainty have been identified. A characterization of those sources of uncertainty and evaluation of the generic sources of uncertainty has not yet been completed however. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-E3	ESTIMATE the uncertainty interval of the overall CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G9, IEC13), taking into	The QU.1 (input) and QU.2 (results) notebooks do not include a parametric uncertainty analysis. Although QU.1 does note that the basis event data (BED) file contains uncertainty distribution data and the basic event uncertainty data in the parameter file is documented in	Not Significant. The parametric uncertainty analysis has been drafted and documented in notebook QU.3, which is currently undergoing acceptance review. The parametric uncertainty analysis has been performed with correlated basic events in order to reflect "state-of-knowledge" dependencies. This is

Gap	Description	Self Assessment	Impact on the RI-ISI Application
	account the "state-of-knowledge" correlation.	the data notebooks (section 2.5), and that uncertainty analyses can be performed on the equation files (section 4.0), there is no such analysis mentioned in QU.2. There are a few basic events in the parameter file (N05A_16C.prm) that do not contain uncertainty distribution data.	judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

Question 2

The supporting requirement (SR), IF-C6 and IF-C8, permits screening out of flood areas based on, in part, the success of human actions to isolate and terminate the flood. The endorsed RI-ISI methods require determination of the flood scenario with and without human intervention which corresponds to the capability category III, i.e., scenarios are not screened out based on human actions. Therefore a category III analysis would be acceptable. To provide confidence that scenarios that might exceed the quantitative CDF and LERF guideline are identified, please describe how credit is given to human actions if the current application analysis does not meet Capability Category III for these supporting requirements.

Dominion Response

Floods were not screened out based on the ability of human actions to isolate or mitigate a flood in the North Anna PRA model. The model meets Capability Category III for IF-C6 and IF-C8, which is appropriate for this application.

NDE Branch

Question 1

Table IWB-2500-1 of ASME, Section XI, 2001 Edition with 2003 Addenda requires volumetric and/or surface examination of all Category B-F or B-J Pressure Retaining Dissimilar Metal Welds greater than NPS 1. Based on recent findings of primary water stress corrosion cracking (PWSCC) in Alloy 82/182 dissimilar metal welds the staff would like more information on your inspection plans for these welds in the 4th Interval ISI Plan for NAPS Units 1 & 2.

Describe the inspection plan of Alloy 82/182 dissimilar metal welds greater than NPS 1 in the 4th Interval ISI Plan for NAPS Units 1 and 2 (e.g., are these welds included in the number of welds selected for examination in the RI-ISI program, how many of these welds are selected for examination, what examination method(s) are being employed, what is the frequency of examination, how is disposition of limited coverage (<90%) examinations handled, etc.).

Dominion Response

PWSCC is an active degradation mechanism (DM) included in the Code Case N-716 (RIS-B) analysis. The checklist criteria for PWSCC is:

a. piping material is Inconel (Alloy 600), and

b. exposed to primary water at temperatures greater than 570 °F, and
c. the material is mill-annealed and cold worked, or cold worked and welded without stress relief.

The method of examination for PWSCC susceptible welds is a volumetric examination using ASME Section XI Figure IWB-2500-8(c). The volume shall be increased by enough distance, approximately 1/2 inch, to include each side of the base metal thickness transition or counterbore transition.

Two areas at North Anna were recognized as susceptible to PWSCC in the RIS-B analysis: 1) Unit 1 Steam Generator hot leg nozzles, and 2) the pressurizer nozzle to safe-end welds, including those for the surge line, spray line and safety and relief valve lines for both units. The Unit 2 Steam Generator hot leg nozzles, which contain the Alloy 82/182 material that is susceptible to PWSCC were inlaid with Alloy 52 during preservice construction. Therefore, the Alloy 82/182 material has never been exposed to primary grade water.

For Unit 1, seven components were assigned PWSCC susceptibility and four (57%) were selected for examination. Two Unit 1 components were assigned DMs for both PWSCC and TT (Thermal Transients). Both of these two components have been selected for examination. For Unit 2, four components were assigned PWSCC susceptibility and one (25%) was selected for examination. Two Unit 2 components were assigned both PWSCC and TT DMs and both were selected for examination. By the Code Case criteria a minimum of 25% of the DMs or combination of DMs should be selected for examination per interval. These selections will be examined once per interval. In this manner, the Class 1 Alloy 82/182 dissimilar metal welds were included in the population analyzed by the Code Case N-716 application to make component selections for examination.

All of the dissimilar metal welds on both Unit 1 and 2 pressurizers have been overlaid with PWSCC resistant material to reinforce the structural integrity. Relief Request NDE-005 for North Anna Unit 1 Interval 4 was developed to address the inspection method and frequency for the pressurizer overlays and was approved by the NRC in a letter dated September 28, 2009 (ML092530274). Dominion plans to inspect the Unit 1 overlaid welds in accordance with this relief request at this time. The same sampling selection of 25% is required by the relief request, so the number of inspections is consistent with the RIS-B approach. However, Code Case N-716 does not address welds that have been overlaid. The inspection techniques and all aspects of the relief request (including determination of additional exams and evaluating indications) will be followed for Unit 1 overlaid selected examinations.

Dominion anticipates Code Case N-770, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds

Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities" will be incorporated by the NRC into the next rule change of the Federal Register by May 2011. This new rule should be effective in time to establish the examination rules for Unit 2 pressurizer overlaid weld inspections. If incorporation of Code Case occurs as anticipated, Dominion will then use the requirements of Code Case N-770 to govern inspection requirements for welds fabricated with Alloy 82/182 material and withdraw the Unit 1 relief request. The inspection selections determined by the RIS-B Program will remain unchanged. However, the guidance of this Code Case will determine the examination methods. If needed, a relief request similar to Unit 1 will be submitted for the Unit 2 pressurizer overlaid Alloy 600 welds, to address inspection method and technique.

North Anna currently has an Augmented Plan that addresses MRP-139, "Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline." Examinations must be performed on Alloy 600 welds until the welds have been mitigated. Currently, the Unit 1 Steam Generator hot and cold leg nozzle to safe end welds must be examined as follows:

- One hot leg nozzle weld: Bare metal visual inspection every refueling outage, UT every period

- One cold leg nozzle weld: Bare metal visual inspection every refueling outage, UT every five years

Code Case N-722 was incorporated into the last publication of the Code of Federal Regulations and was implemented by January 2009 at North Anna. The Code Case requires a visual bare metal (VE) inspection on unmitigated, Class 1, Alloy 600 welds. Steam Generator hot leg nozzle-to-pipe-welds must receive a VE inspection every refueling outage and the Steam Generator cold leg nozzles must be VE inspected once per interval.

If any RIS-B selections are made on welds that have not been mitigated (i.e., Steam Generator hot legs) they will be volumetrically examined in accordance with Code Case N-716 using ASME Section XI Figure IWB-2500-8(c). The volume shall be increased by enough distance, approximately 1/2 inch, to include each side of the base metal thickness transition or counterbore transition. Dominion has used the phased array UT technique previously on the Steam Generator nozzle welds to achieve maximum coverage and obtain acceptable results and plans to continue to use the phased array technique on the Steam Generator welds.

Part of the initial selection process for determining RIS-B examinations is to choose components that are known to meet full coverage requirements. If limited examinations (coverage 90% or less) do occur, Dominion will address the limitations by a relief request in accordance with 10 CFR 50.55a(g)(5)(iii).

To summarize, presently there are several drivers to inspect Alloy 600 welds susceptible to PWSCC: Code Case N-722, MRP-139 and the proposed Risk Informed Inservice Inspections. North Anna has programs in place to address each of these independently. The RIS-B analysis was performed without consideration of any other Programs for inspecting Alloy 600 welds; such as MRP-139 and Code Case N-722. The analysis of the RIS-B Program did not credit exams scheduled to meet N-722 or the Augmented Program (MRP-139) in any manner to reduce the need for inspections. If welds have been overlaid they are still noted in the RIS-B Program as susceptible to PWSCC and will be selected for examination as required. The technique for examining the overlaid welds will follow criteria of either an accepted relief request or Code Case N-770 if incorporated into the Code of Federal Regulations. Examination overlap may occur and credit for one weld exam may be taken for multiple programs if the examination specifications/requirements for each program credited are met.

NRC Question 2

Section 3.3 of the February 23, 2010 submittal states that, "In contrast to a number of RI-ISI Program applications where percentage of Class 1 piping locations selected for examination has fallen substantially below 10%, Code Case N-716 mandates that 10% be chosen." Immediately below this paragraph a brief summary is provided showing the number of welds in Class 1, 2 and non-class systems along with the number of welds selected for examination for NAPS Units 1 & 2. According to this summary the number of Class 1 welds selected for examination on Unit 2 is significantly less than 10% of the total number of Class 1 welds. Please explain this discrepancy.

Dominion Response

At North Anna, the Class 1 boundaries have been unnecessarily extended beyond the second isolation valve from the reactor pressure vessel. This was done to coordinate the "Q" Quality boundary designations that were made during system design and construction.

In determining the High Safety Significant (HSS) components that are subject to selection for examination, Code Case N-716 defines HSS welds as:

(1) Class 1 portions of the reactor coolant pressure boundary (RCPB), except as provided in 10 CFR 50.55a(c)(2)(i) and (c)(2)(ii).

10CFR50.55a(c) Reactor *Coolant Pressure Boundary* (2)(ii) states "The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only."

Part (2) of the Code Case further defines HSS components:

(2) applicable portions of the shutdown cooling pressure boundary function. That is, Class 1 and 2 welds of systems or portions of systems needed to utilize the normal shutdown cooling flow path either:

(a) as part of the RCPB from the reactor pressure vessel (RPV) to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds; or
(b) other systems or portions of systems from the RPV to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds.

Section 3.1 of the February 10, 2010 submittal reiterates this information.

Based on the definitions in Code Case N-716, 263 of Unit 2 Class 1 Safety Injection welds are Low Safety Significant (LSS), 513 of Class 1 Charging welds are LSS. For Unit 1, 38 of Class 1 Safety Injection welds are LSS and 77 Class 1 Charging welds are LSS. These welds are not required to be included in the HSS population, but are included in the total Class 1 weld count in the table of Section 3.3 of the February 10, 2010 submittal.

The statement in Section 3.3 when addressing the unique classification definitions at North Anna, would be better stated, "In contrast to a number of RI-ISI Program applications where the percentage of HSS Class 1 piping locations selected for examination has fallen substantially below 10%, Code Case N-716 mandates that 10% be chosen." At most plants the total number of Class 1 components will be HSS; however, at North Anna Units 1 and 2, they are not.

Dominion's engineering document "Risk Informed Inservice Inspection Program for NAPS 1 and 2 4th Intervals, Code Case N-716 Based" was written to support the RI-ISI submittal. Enclosure 1 contains pages of that Engineering document. The Weld Count tables show the LSS and HSS totals. Using the RIS-B application, 10% of the HSS welds are required to be examined, which is 10 % of 1433 for Unit 1 and

10% of 1528 for Unit 2. The total number selected shown in the Section 3.3 table of the Submittal, 178 for Unit 1 and 183 for Unit 2 is correct.

NRC Question 3

Also the total number of welds shown in the summary in Section 3.3 for Unit 2 does not agree with the "Weld Count" column total value shown in Table 3.1b of the February 23, 2010 submittal. Please explain this discrepancy.

Dominion Response

Table 3.1b for NAPS 2 was not correct and contained erroneous totals for Low Safety Significant welds. The following are the correct values for total LSS welds: Main Steam-- 171 versus 160, Residual Heat - 141 versus 139, Safety Injection - 736 versus 734, Quench Spray - 167 versus 165, Recirculation Spray - 88 versus 86. Total LSS welds should be 2244 versus 2234 and the total weld count should be 3772 versus 3762. Enclosure 2 to this letter contains a corrected table. Please replace Table 3.1b in the original submittal with the updated information presented in Enclosure 2. The total weld counts and number of selections in the Section 3.3 table of the submittal is correct.

During the development of this response we discovered a typographical error. On Table 3.3a, for the RC system TT (Thermal Transients) should pair with PWSCC, not TASCS (Thermal Stratification, Cycling and Striping). Enclosure 3 is the corrected Table 3.3a for NAPS 1. Please replace this Table in the original submittal.

Enclosure 1

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

System	DM Welds (25%)			HSS (10%)	RCPB (10%)			RCPBu (2/3 of RCPB)			BER (10%)		
	Total	Selected	% DM	Total HSS %	Total	Selected	%	Total	Selected	% RCPB	Total	Selected	%BER
CH - Charging	18	5	27.8%	16.0%	319	51	16.0%	59	49	96.1%	0	0	0.0%
FW - Main Feedwater	0	0	0.0%	11.3%	0	0	0.0%	0	0	0.0%	43	10	23.3%
MS - Main Steam	0	0	0.0%	20.0%	0	0	0.0%	0	0	0.0%	30	6	20.0%
RC - Reactor Coolant	77	22	28.6%	10.9%	570	62	10.9%	452	62	100.0%	0	0	0.0%
RH - Residual Heat	0	0	0.0%	12.9%	31	4	12.9%	9	2	50.0%	0	0	0.0%
SI - Safety Injection	9	3	33.3%	11.4%	368	42	11.4%	0	0	0.0%	0	0	0.0%
Total	104	30	28.8%	178	1288	159	12.3%	520	113	71.1%	73	16	21.9%
Total Check	104	30	28.8%	12.4%	1288	159	12.3%	520	113	71.1%	73	16	21.9%

Total Selected	178
Total Section XI Inspe	294

Weld Count

System	Total	HSS	LSS
CH- Charging	811	319	492
FW - Main Feedwater	115	115	0
MS - Main Steam	199	30	169
RC - Reactor Coolant	570	570	0
RH - Residual Heat	168	31	137
SI - Safety Injection	860	368	492
QS - Quench Spray	143	0	143
RS - Recirc Spray	73	0	73
Total	**2939**	**1433**	**1506**

System	DM Welds (25%)			HSS (10%)	RCPB (10%)			RCPBu (2/3 of RCPB)			BER (10%)		
	Total	**Selected**	**% DM**	**Total HSS %**	**Total**	**Selected**	**%**	**Total**	**Selected**	**% RCPB**	**Total**	**Selected**	**%BER**
CH - Charging	20	6	30.00%	11.90%	420	50	11.90%	58	47	94.00%	0		
FW - Feedwater	0	0	0.00%	13.27%	0	0	0.00%	0	0	0.0%	44	15	34.1%
MS - Main Steam	0	0	0.00%	18.18%	0	0	0.00%	0	0	0.0%	26	6	23.1%
RC - Reactor Coolant	78	27	34.62%	11.87%	573	68	11.87%	472	68	100.0%	0	0	0.0%
RH - Residual Heat Removal	0	0	0.00%	13.79%	29	4	13.79%	10	2	50.0%	0	0	0.0%
SI - Safety Injection	9	3	33.33%	11.11%	360	40	11.11%	0	0	0.0%	0	0	0.0%
Total	107	36	33.64%	183.00	1382	162	11.72%	540	117	72.2%	70	21	30.0%
Total Check	107	36	33.64%	11.98%	1382	162	11.72%	540	117	72.2%	70	21	30.0%

Total Selected	183
Total Section XI Inspected	279

Weld Count

System	Total	HSS	LSS
CH - Charging	1340	420	920
FW - Main Feedwater	134	113	21
MS - Main Steam	204	33	171
RC - Reactor Coolant	573	573	0
RH - Residual Heat Removal	170	29	141
SI - Safety Injection	1096	360	736
QS - Quench Spray	167	0	167
RS - Recirc Spray	88	0	88
Total	**3772**	**1528**	**2244**

Enclosure 2

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Table 3.1b
N-716 Safety Significance Determination: NAPS2

System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	>1E-6^{CDF} >1E-7^{LERF}	High	Low
CH – Charging	420	✓					✓	
	920							✓
FW - Main Feedwater	44				✓		✓	
	69			✓			✓	
	21							✓
MS - Main Steam	33				✓		✓	
	171							✓
RC - Reactor Coolant	573	✓					✓	
RH - Residual Heat	29	✓	✓				✓	
	141							✓
SI - Safety Injection	360	✓					✓	
	736							✓
QS - Quench Spray	167							✓
RS- Recirculation Spray	88							✓
SUMMARY RESULTS FOR ALL SYSTEMS	1353	✓					✓	
	77				✓		✓	
	69			✓			✓	
	29	✓	✓					
	2244							✓
TOTALS	**3772**						**1528**	**2244**

Enclosure 3

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Table 3.3a
N-716 Element Selections: NAPS1

System(1)	Selections	HSS(2)	DMs(3)	RCPB(4)	RCPB^IFIV(5)	RCPB^OC(6)	BER(7)
CH	**Required**	32 of 319	TT 5 of 8	32 of 319	37	n/a	n/a
	Made	51	TT 5	51	49	n/a	n/a
FW	**Required**	12 of 115	n/a	n/a	n/a	n/a	5 of 43
	Made	13	n/a	n/a	n/a	n/a	10
MS	**Required**	3 of 30	n/a	n/a	n/a	n/a	3 of 30
	Made	6	n/a	n/a	n/a	n/a	6
RC	**Required**	57 of 570	TASCS, TT 4 TASCS 11 of 41 TT 3 of 11 PWSCC 2 of 7 PWSCC, TT 1 of 2	57 of 570	41	n/a	n/a
	Made	62	TASCS, TT 7 TASCS 5 TT 4 PWSCC 4 PWSCC, TT 2	62	62	n/a	n/a
RH	**Required**	3 of 31	n/a	3 of 31	3	n/a	n/a
	Made	4	n/a	4	2	n/a	n/a
SI	**Required**	37 of 368	IGSCC 2 of 6 TT, IGSCC 1 of 3	37 of 368	0	n/a	n/a
	Made	42	IGSCC 2 TT, IGSCC 1	42	0	n/a	n/a
TOTAL	**Made**	**178**	**30**	**159**	**113**	**n/a**	**16**

Notes

(1) Systems are described in Tables 3.1a and 3.1b.
(2) High Safety Significant
(3) Degradation Mechanisms No more than 10% of HSS piping welds are required to be selected for examination. DM selections may be reduced to meet this requirement.
(4) Reactor Coolant Pressure Boundary
(5) For RCPB^IFIV (Reactor Coolant Pressure Boundary inside first isolation valve) 2/3 requirement is for total of RPCB^IFIV and is not required to be met per system.
(6) Reactor Coolant Pressure Boundary outside containment
(7) Break Exclusion Region