VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 27, 2000

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20005

Serial No. 00-229A
NL&OS/GDM R2
Docket No. 50-281
License No. DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT 2
RISK-INFORMED INSERVICE INSPECTION PROGRAM
REQUEST FOR ADDITIONAL INFORMATION

In a letter dated April 27, 2000 (Serial No. 00-229), Virginia Electric and Power Company submitted a Risk-Informed Inservice Inspection (RI-ISI) Program for Surry Power Station, Unit 2. During the NRC's review of the submittal, the staff identified a need for additional information to facilitate their review. The staff provided their questions on August 3, 2000, and we responded to the questions during a conference call on August 8, 2000. At the conclusion of the conference call, the NRC requested that we formalize our responses to the staff's questions in a letter. Subsequent to the conference call, the NRC provided three additional questions regarding socket weld examinations. Our responses to both sets of questions are provided in the attachment.

If you have any questions or require additional information, please contact us.

Very truly yours,

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ATTACHMENT

RISK-INFORMED INSERVICE INSPECTION (RI-ISI) PROGRAM
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SURRY POWER STATION UNIT 2

Attachment

Response to NRC Request for Additional Information - RI-ISI Program
Surry Power Station Unit 2

NRC Question No. 1

The licensee's letter of April 27, 2000 states that it is the licensee's intent to complete 100% of the RI-ISI locations over the current third inspection interval by either the Section XI ISI program or by the proposed RI-ISI program. Please clarify the intent, and also verify whether Surry Unit 2 would be prepared to complete the required percentage of inspections for the second inspection period of the third inspection interval, as specified in the table IWB-2412-1 of the 1989 Edition of ASME Code, Section XI.

Virginia Power Response

As indicated in our letter and associated relief request dated December 10, 1999 (Serial No. 99-518), which was approved by the NRC in a Safety Evaluation Report dated April 19, 2000, and our program submittal dated April 27, 2000, it is our intention to complete the RI-ISI program (100%) by the end of the third inspection interval. Credit would be given to locations already examined by the current Section XI program if they correspond to RI-ISI locations. Since only one outage remains for the Surry Unit 2 second inspection period, it is not our intent to meet the percentage requirements of the ASME Code for that period, but rather compensate for the difference in the third period.

NRC Question No. 2

In Tables 5-1 and 5-2 regarding Class 1 structural element selection results and comparison of inspection requirements between the Section XI program and the RI-ISI program in Surry 1 and Surry 2, please explain (1) how many socket welds are included in the B-J welds at Surry 2, (2) how many butt welds are included in the B-J welds at Surry Unit 2, and (3) how to tie the 84% NDE reduction in Surry Unit 2, as claimed in the Summary of Results, with figures presented in Table 5-1.

Virginia Power Response

(1) The current number of ASME Section XI socket welds (B-J) is 639 of which 161 are examined using the surface examination method (25.2%). The RI-ISI program proposes visual (VT-2) examinations.

(2) The current number of ASME Section XI butt welds is 18 (B-F) and 459 (B-J) of which 18 (100%) of the B-F welds are examined and 123 (26.8%) of the B-J welds are examined. The RI-ISI program proposes to volumetrically examine fifty butt

welds. The selected population includes six former B-F welds and 44 B-J welds or approximately 10.5% of the butt weld population. Note the counts for items (1) and (2) above do not include long seam welds, as they are examined in conjunction with the circumferential weld.

(3) Table 5-1 indicates that the current ASME Section XI program examines eighteen B-F locations and 295 B-J locations with NDE. Since the RI-ISI program proposes examination of 50 locations for NDE, that represents an 84% reduction from the current ASME Section XI total.

NRC Question No. 3

In Relief Request R-1, the basis for relief indicates that certain socket weld connections are identified as High Safety Significant (HSS) and require volumetric examination for their postulated failure mechanism, such as potential thermal fatigue damage mechanism either caused by a postulated temperature stratification or as a default mechanism with no assumed active mechanism occurring. Please (1) identify the piping subjected to postulated temperature stratification, (2) clarify the default mechanism, (3) discuss and justify why the requested alternative examination for HSS socket welds is VT-2 instead of surface examination as required in Table IWB-2500-1 and Table 4.1-1 of the WCAP for socket welds in Examination Category B-J. (Note that the recently revised version of the Code Case N-577-1 has not yet been reviewed by the NRC.)

Virginia Power Response

(1) The piping segments (socket welded connections) affected by potential temperature stratification were segments ECC-008, 009, 010 (high pressure cold leg safety injection). These segments are made up of both butt welds and socket welds. The butt welds will be volumetrically examined in the RI-ISI program.

(2) The WOG methodology approved for WCAP-14572 Rev. 1-NP-A does not have a postulated "no active failure mechanism" category. The default postulated mechanism is thermal fatigue (normal stresses from heat-ups and cooldowns, etc.) when nothing else has been postulated. The examination default is a volumetric examination of the location. This would be applied to the default failure mechanism locations for Region 1(b) and 2. The piping segments (socket welded connections) affected by the potential default failure mechanism were segments ECC-004 (hot leg high pressure safety injection header), ECC-011, 012, 013 (cold leg safety injection 3/4" line) and RC-40 (reactor coolant letdown).

(3) The ASME Code through Code Cases N-577-1 and N-578-1 is on record as endorsing the substitution of visual (VT-2) exams for volumetric exams on socket welds. A surface exam on a socket weld on a ten-year frequency is not likely to

identify any ID originating damage mechanism and is considered an unnecessary radiation exposure burden. A frequent visual examination (VT-2) focused on the area of concern is the best alternative as proposed by the Code. The industry (NEI/EPRI) met with the NRC on August 29, 2000 to discuss RI issues. The VT-2 exam for socket welds was discussed and the proposed substitution was again endorsed by the industry. It was noted during the meeting that the EPRI-MRP thermal fatigue task group was due to issue a formal report in 2001. The report will be reviewed by Virginia Power for any impact on the RI-ISI program and will be considered as new information with regard to socket welds and that postulated damage mechanism.

NRC Question No. 4

How did you estimate socket weld failure likelihoods? What type of POD (probability of detection) did you use for socket welds? Did you increase or decrease the POD when changing from PT and VT in ASME Section XI to only VT exams in RI-ISI?

Virginia Power Response

The socket weld failure likelihood was determined by using the Win-SRRA software code (Windows version of SRRA model described in WCAP-14572 Rev. 1-NP-A Supplement 1) in conjunction with the failure probability sub-panel and expert panel review. The values used were compared to the postulated failure mechanism and to the values of the other segments in determining their credibility. The risk ranking of segments was performed with the failure probability assuming no Section XI ISI credit. Section XI ISI credit (failure probability used) was used only in the change in risk calculation. The failure probability crediting ISI (change in risk calculation) assumes the same probability of detection for visual and surface exams, which is the default low setting in the Win-SRRA software code.

NRC Question No. 5

The WCAP calls for 100% inspection in the Region 1 elements exposed to degradation mechanism, and a Perdue sampling (which tends to default to 1 inspection) in the remaining elements in the HSS segment not exposed to degradation mechanisms. Your submittal states that "the number of locations to be inspected in a HSS segment was determined using the Westinghouse statistical method." Please explain this apparent discrepancy?

Virginia Power Response

The use of the "Westinghouse statistical model" term was intended as a general reference to the selection process described in the WCAP-14572 Rev. 1-NP-A. Surry Unit 2 followed the WCAP selection criteria including the use of the Perdue sampling for

Regions 1(B) and 2, and either 100% selection or the existing augmented program for Region 1(A).

NRC Question No. 6

The CDF/LERF from the PRA used for Surry Unit 1 RI-ISI was 7.2E-5/yr and 1.1E-5/yr respectively. The current PRA estimates CDF/LERF as 3.8E-5/yr and 2.7E-6/yr respectively. It is our understanding that one PRA is used for the nominally identical units, is this correct? Please briefly describe the changes to the PRA that resulted in these rather large changes to the results.

Virginia Power Response

One PRA model was used to develop the Risk-Informed Inservice Inspection (RI-ISI) Programs for both Surry Units 1 and 2, since the units are nominally identical to each other. However, two different versions of the same PRA model were used to develop the RI-ISI programs for each unit. The PRA model that updated the Individual Plant Examination (IPE) in August 1993 was used to develop the RI-ISI program for Surry Unit 1 (CDF and LERF as 7.2E-5/yr and 1.1E-5/yr, respectively). For the Surry Unit 2 RI-ISI program, the latest version of the PRA model was used (CDF and LERF as 3.8E-5/yr and 2.7E-6/yr, respectively), which had been updated in June 1998 (with three year average unavailability). During the PRA model updating process in 1998, more than a dozen changes were made to the previous version of the PRA model that was used to develop the Unit 1RI-ISI program and that resulted in changes in the CDF and LERF values. Some of the major changes to the model included revising success criteria based upon new calculations for Emergency Switchgear Room (ESGR) heatup and Feed & Bleed. Other changes include adding emergency power models for the Station Blackout (SBO) diesel generator and shared emergency diesel generator auctioneering.

NRC Question No. 7

Did you evaluate the results according to all the criteria on page 214 and 215 of the WCAP? How does the CH system, which has a relatively high CDF and LERF contribution in Table 3.5-1, and currently has 70 B-J welds inspected, reduced to 6 VT-2 inspections and yet have no impact on change in risk.

Virginia Power Response

There are fourteen (14) Class 1 segments in CH (Chemical and Volume Control System). The first three segments (CH-1, CH-2, and CH-3) are reactor coolant pump seal injection lines (2 inches in diameter). Quantitatively the segments were high safety significant (HSS), and the Expert Panel voted these as HSS segments. Segments CH-4, CH-5, and CH-6 are the reactor coolant pump seal injection lines that are connected to the segments CH-1, CH-2, and CH-3, respectively. Quantitatively these segments were low safety significant (LSS) but were determined HSS by the Expert

Panel since each segment is separated with a single untested check valve from the adjoining HSS segment. Segments CH-7 through CH-11 are in the charging system, and these segments have no failure effects. The remaining three segments (CH-12, CH-13, and CH-14) are the seal bypass lines. Quantitatively these segments were of medium safety significance (LERF only), but on the low end of medium. The Expert Panel unanimously voted the segments LSS. As a result, only six out of fourteen segments were determined to be HSS. The failure mechanism for these six segments is vibratory fatigue; therefore, only VT-2 inspections are required. Evaluation of the change in risk between ASME Section XI ISI and RI-ISI was performed in accordance with the criteria on pages 214 and 215 of WCAP-14572, Revision 1-NP-A. Initial CDF/LERF comparisons were made between the proposed RI-ISI program (based on the Expert Panel's selections) and the current ASME Section XI ISI program to evaluate the effect of the RI-ISI program on risk. The initial comparison results revealed a slight increase in risk in ACC, RC, and RH systems; however, they were all within the criteria established by the WCAP. The CH system was risk neutral. The dominant system contributor (i.e., system contribution to the total is greater than approximately 10%) for CDF with and without operator action is the RC system (91.4%). The dominant system contributors for LERF with and without operator action are the RC system (89.8%) and the CH system (10%). Although, the CH system contributed 10% of the total LERF for the with and without operator action cases, the change in piping risk was risk neutral. Therefore, only the RC system was examined further for risk changes. After adding two more RC segments (RC-46 and RC-48) to be inspected, the results of the change in risk show that modifying examination requirements from the current ASME Section XI ISI program to the proposed RI-ISI program is an enhancement or at least risk neutral.

NRC Question No. 8

Please clarify the entry under section 3.4 that, "another consideration was whether a segment is addressed by the plant stress corrosion cracking augmented program." Please also provide (1) the total number of welds currently being inspected in the augmented program at Surry 2, (2) the number of these augmented program inspections included in the 50 RI-ISI volumetric examinations in Table 5-1, and (3) the number of VT inspections in the augmented inspections (if any) included in Table 5-1. Were the subsequent changes in risk calculations performed as described on page 213 of the WCAP?

Virginia Power Response

(1) Three piping segments are part of the augmented inspection program for Surry Unit 2. The Technical Specifications for Surry requires that the welds be inspected at three times the frequency of the Section XI program (75% in 10 years). There are approximately 18 welds considered part of the augmented program on these segments.

(2) No augmented inspections are credited in the proposed 50 volumetric examinations for the RI-ISI program.

(3) No VT inspections are associated with the augmented program.

Subsequent changes in risk calculations were performed as described on page 213 of the WCAP and are addressed in our response to Question 7 above.

NRC Question No. 9

According to an ASME Code Case under preparation, elements subject to external chloride stress corrosion cracking (ECSCC) should be examined by surface examination methods. Please justify if any of Surry Unit 2 Class 1 socket weld locations being proposed for VT-2 examination is susceptible to ECSCC.

Virginia Power Response

The proposed Surry Unit 2 Class 1 RI-ISI program did not identify any socket weld locations susceptible to ECSCC. The Class 1 socket weld piping is not located in areas that are subject to an aggressive environment that would promote ECSCC, i.e., there are very low levels of chloride (if any) and moisture is not typically present on the pipe.

NRC Question No. 10

Please justify if any of Surry Unit 2 Class 1 socket weld locations being proposed for VT-2 examination is susceptible to any other outside diameter (OD) initiated degradation mechanism such as thermal fatigue, etc.

Virginia Power Response

The damage mechanisms identified were postulated as inside diameter (ID) initiated, such that a surface examination would not be of significant benefit. The only thermal fatigue that the subject piping would be subject to would be the low cycle fatigue considered in the design, which has a very low probability of occurrence. Plant experience indicates other fatigue damage would be manifested as ID initiated cracking. No other externally driven damage mechanism can be reasonably postulated for this piping.

NRC Question No. 11

If not susceptible to ECSCC/OD initiated thermal fatigue for the scope covered by the proposed RI-ISI program, what other socket weld locations in the plant may be affected by such a mechanism?

Virginia Power Response

Some evidence of ECSCC was discovered on Class 2 piping; however, the piping was greater than 2" in diameter (butt welded). Additionally, the augmented sensitized stainless steel inspection program (Class 1 and Class 2) for Surry does postulate the potential for OD initiated conditions on the sensitized piping. As such, Surry continues to perform augmented examinations as required by Technical Specifications (volumetric and surface examinations as required by the 1989 Section XI Code at 3 times the frequency.)