VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 20, 1998

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

Serial No. 98-421A
NL&OS/GDM R0
Docket No. 50-280
License No. DPR-32

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT 1
REQUEST FOR ADDITIONAL INFORMATION - SUPPLEMENT
RISK-INFORMED INSERVICE INSPECTION (RI-ISI) PILOT PROGRAM

At the NRC's request, a conference call was conducted on September 16, 1998, with representatives of Virginia Electric and Power Company (Virginia Power), Westinghouse Electric Corporation, the NRC and their contractors. The NRC requested the conference call to obtain clarification of two items associated with Virginia Power's Risk Informed Inservice Inspection Pilot Program submittal for Surry Unit 1. The two items requiring clarification are as follows: 1) Virginia Power's selection process for the number of piping segment elements to be inspected in the RI-ISI program when the Perdue Model is not used, and 2) assumptions associated with the performance of automatic containment isolation valves.

Piping Segment Element Selection Process

The discussion primarily concerned the methodology that was used to determine when the Perdue Model was to be used in the selection of an appropriate number of inspection locations per segment, and when the Perdue model was not used, how the number of inspection elements per segment was determined. The segments and inspection elements were included in Table 15-1, "Segments Selected for Examination," which was provided in response to NRC Question No. 15 in our letter dated August 13, 1998 (Serial No. 98-421). The NRC observed that while the text of the Surry submittal and the supporting WCAP-14572 seems to indicate heavy reliance on the Perdue Model for inspection element selection, Table 15-1 includes a large number of segments where the Perdue Model was not used. The reason for this difference is that the Perdue Model is intended for use with piping that is considered highly reliable so that a statistically relevant sample can be selected to provide a reliable determination of the condition of the piping. As discussed in WCAP-14572, in cases where the piping in question has an active degradation mechanism, particularly one in which there is an ongoing augmented inspection program in place, the use of the Perdue Model for

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element selection may be inappropriate. In other cases, limitations on the nondestructive evaluation techniques that can be applied to the piping segment limit the relevance of application of the Perdue Model. In these cases, the expert subpanel applied other rationales for the selection of an appropriate number of inspection elements.

The following segments from Table 15-1 are examples of these types of situations:

CH-008, 009, and 010 – These three segments within the charging system are small bore, socket welded piping segments which supply seal injection water to the reactor coolant pump seals. The predicted failure mechanism is high cycle fatigue due to pump vibration. The examination technique required by Code Case N-577 is a VT-2 exam at each refueling outage. Since the VT-2 exam involves inspection of the whole segment for leakage at pressure, tabulation of the exact number of welds per segment and application of the Perdue Model was not deemed necessary. This would be the case for any segment where VT-2 is the appropriate inspection technique. The additional NDE noted in Table 15-1 is directed by the subpanel and is over and above the requirements of the Code Case and the WCAP.

FW-001, 002, 003, 004, 005, 006, and 007 – These non-safety related feed water system segments are subject to flow assisted corrosion (FAC) and are included in an augmented inspection program. Because they are included in an augmented program that dictates the number and location of examinations, the Perdue model was not used to select the number of examination locations for the primary failure mechanism (FAC). A secondary failure mechanism was postulated (thermal fatigue) and examination locations, one per segment, were selected to improve the delta risk of RI-ISI versus Section XI ISI. The Perdue Model was not used in cases where examination locations were selected for delta risk or defense in depth considerations because these are over and above the requirements of the Code Case.

SW-044, 045, 046, 047, and 054 – These service water segments are fabricated of copper/nickel material which is not a material that can be modeled by the SRRA code used for Surry Unit 1. These segments direct service water to and from the charging pump intermediate seal coolers. The segments were originally rated as low safety significance but were subsequently recategorized as high safety significance by the Expert Panel because of its sensitivity to the possibility of indirect effects. Because 1) the piping is considered highly reliable, 2) the postulated failure mechanism is thermal fatigue by default (although thermal cycles are practically nonexistent), and 3) the SRRA code could not be used to calculate a failure probability, which is a necessary input to the Perdue Model, the Model was not used to select examination locations. The subpanel believed that one examination location per segment would be representative of the balance of the highly reliable, low safety significant segments.

Performance Consideration of Automatic Containment Isolation Valves

The NRC questioned whether Virginia Power took credit for the operation of automatic containment isolation valves for establishing segment boundaries. We did assume containment isolation valves operated as required for affected segments. Furthermore, the system descriptions presented to the Expert Panel provided information relative to segments that might have an effect on large early release frequency (LERF) probabilities. The Expert Panel discussions routinely dealt with automatic isolation considerations. For example, charging system segment CH-005, which numerically rated a low safety significance, was moved to a high safety significance by the Expert Panel because of the importance of the segment relative to maintaining containment integrity. Containment isolation valve integrity was explicitly considered through the Expert Panel meetings, and, when the conclusions were significant, it was documented in the meeting minutes. Samples of these meeting minutes were previously provided to the NRC during the July 23, 1998 meeting at NRC headquarters.

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

Very truly yours,

L. N. Hartz
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Commitment Summary: No new commitments are provided in this correspondence.

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