VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 30, 1980

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Stephen A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 802
NO/FHT:jmj
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Dear Mr. Denton:

SURRY POWER STATION UNITS 1 AND 2
REACTOR VESSEL FLANGE SEAL RING

In response to a request from Mr. Don Neighbors of your staff, we are forwarding the following updated status of the subject issue.

Our letter of May 5, 1978, stated that the reactor vessel seal ring issue, which is concerned with the seal ring becoming a missile due to reactor cavity pressurization during certain LOCA events, would be resolved concurrent with the Asymmetric LOCA Loads issue. It was our intention to identify the reactor cavity pressure transient due to a double-ended guillotine (DEG) rupture of a reactor vessel nozzle within the reactor cavity. An analysis was then to be performed to determine the effect of the expected pressure transient on the seal ring itself.

Concurrent with the Asymmetric Loads evaluations, the Westinghouse Owners Group, of which we are a member, has been engaged in a Mechanistic Fracture Evaluation. This evaluation has addressed the mechanics of the propogation of postulated through-wall reactor coolant pipe cracks. The results of the evaluation of reactor coolant pipe base metal behavior were reported in WCAP-9558, Revision 1, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack." This report was submitted to the NRC by Westinghouse letter NS-TMA-2265, dated June 30, 1980, and referenced on Surry's dockets by our letter Serial No. 612, dated July 14, 1980.

This report and the NSAC/EPRI Technical Memorandum submitted to the NRC on October 19, 1979, in a letter from John E. Ward (Chairman, AIF Committee on Reactor Licensing and Safety) to Harold R. Denton, have determined, by diverse and independent analyses and experimental results, that the probability of high energy line breaks in reactor piping systems, both austenitic and ferritic, is extremely small. In addition, the consequence of unanticipated, slow crack growth due to fatigue, corrosion fatigue, or stress corrosion cracking is likely to be relatively slow leakage. The analyses specifically determined that very large cracks are required to initiate ductile fracture in nuclear piping under normal loadings and that unstable crack extension is unlikely to occur, and the openings of through-wall cracks are small.

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In addition to this evaluation of reactor coolant pipe base metal fracture mechanics, and at the request of your staff, the Owners Group has recently sponsored a supplemental evaluation of piping circumferential weld metals. The results of this study, to be published as WCAP-9787 in October, 1980, indicate that, for the circumferential welds in primary piping at Surry Power Station, the weld metal will behave similarly to the base metal with regard to crack growth.

These results support the conclusion that a DEG break in a reactor coolant system pipe without any prior indication of substantial leakage is unrealistic and need not be considered as a basis for plant design or modification. This rationale has been proposed to, and accepted by, the NRC Staff as justification for continued operation of Surry 1 and 2 without installation of pipe break restraints pending completion of the Fracture Mechanics Evaluation. It is expected, based on the encouraging results to date, that installation of pipe break restraints will not be required and that the DEG break will no longer be considered as a basis for plant design or modification.

For the reasons stated above, reactor cavity pressure transients due to vessel nozzle breaks will be insignificant in terms of loadings on the seal ring. The maximum break size of a few square inches equivalent diameter, as shown in the Mechanistic Fracture Evaluation, does not pose a threat to the integrity of the seal ring.

We agree with the NRC Staff assessment, "that the probability of a pipe break resulting in substantial transient loads on the vessel support system or other structures is acceptably small because: 1) the break of primary concern must be very large, 2) it must occur at a specific location, 3) the break must occur essentially instantaneously, and 4) the welds are currently subject to inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI." This is further justification for delaying analysis of and/or modifications to the reactor vessel flange seal ring until the Fracture Mechanics Evaluation can be fully reviewed by the NRC.

Very truly yours,

BR Sylvia

B. R. Sylvia
Manager - Nuclear
Operations and Maintenance

cc: Mr. James P. O'Reilly