VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

August 5, 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Lester S. Rubenstein, Director
PWR Project Directorate No. 2
Division of PWR Licensing-A
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 86-477
NO/DJV/acm
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
REDESIGN OF REACTOR COOLANT PUMP
AND STEAM GENERATOR SUPPORTS

By letter dated April 30, 1986, Virginia Electric and Power Company requested an amendment to the Operating Licenses for Surry Power Station Units 1 and 2 permitting the redesign of the reactor coolant pump and steam generator supports in accordance with the requirements of General Design Criterion (GDC) 4. As you are aware, GDC 4 was recently revised to permit the application of "leak-before-break" technology for excluding from the design basis the dynamic effects of postulated pipe ruptures in primary coolant loop piping in pressurized water reactors. Our amendment request was approved by the NRC in license amendment no. 108, dated June 16, 1986. The purpose of this letter is to inform you of our intention to request a similar amendment for North Anna Power Station Units 1 and 2 and to provide our schedule for submittal of the request and related information.

The scope of our pending request is discussed in the attachment. Your approval of this request will permit elimination of certain large bore snubbers which are now required solely to mitigate a pipe rupture event. The technical basis for our request is also discussed in the attachment. Our formal request for license amendment including the detailed reports supporting our technical conclusions are scheduled for submittal in October, 1986.

If you have any questions on this matter, please contact us.

Very truly yours,

W. L. Stewart

Attachment

A001
1/1

cc: Dr. J. Nelson Grace
Regional Administrator
NRC Region II

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

Mr. Leon B. Engle
NRC North Anna Project Manager
PWR Project Directorate No. 2
Division of PWR Licensing-A

<u>ATTACHMENT</u>

REDESIGN OF REACTOR COOLANT PUMP AND STEAM GENERATOR SUPPORTS

Scope and Justification for Request

ATTACHMENT

REDESIGN OF REACTOR COOLANT PUMP AND STEAM GENERATOR SUPPORTS
SCOPE AND JUSTIFICATION

SCOPE OF REQUEST

Based on the recent revision to GDC-4 (Reference 1), the Virginia Electric and Power Company intends to request approval for redesign of the reactor coolant pump and steam generator supports at North Anna Units 1 and 2. The new GDC-4 eliminates the need for consideration of postulated breaks in the reactor coolant system primary loop piping and its associated dynamic and other effects such as pipe whip, jet impingement, assymmetric pressure loading, and primary component sub-compartment pressurization. Approval of this request will also allow us to eliminate certain large bore snubbers which are now required solely to mitigate a pipe rupture event. Specifically the redesign of the support system for the main reactor coolant piping and components will allow us to:

(1) Eliminate two snubbers per loop at the steam generator upper support ring and replace them with rigid links.

(2) Eliminate two snubbers per loop which are parallel to the reactor coolant hot leg at the steam generator lower support.

(3) Eliminate two snubbers per loop which are parallel to the reactor coolant cold leg at the reactor coolant pump support.

(4) Eliminate two of four snubbers per loop which are crossover restraints between the reactor coolant pump and the steam generator.

Granting this request would not affect:

- Emergency Core Cooling System (ECCS) design basis.
- Reactor building and compartment design basis.
- Equipment qualification basis.
- Engineered safety feature systems response.

JUSTIFICATION FOR REQUEST

This request will be based upon the use of advanced fracture mechanics technology as applied to primary system piping in Westinghouse Electric Corporation topical reports (References 2, 3, and 4 with approval by the USNRC in Reference 5), and by WCAP-11163 and 11164 (Reference 6) in preparation specifically for North Anna Units 1 and 2.

Generic Letter 84-04 provided the NRC staff Safety Evaluation Report for analysis of materials submitted for a group of utilities operating PWR's to resolve generic issue A-2. The staff evaluation concluded that provided certain conditions were met, an acceptable technical basis exists so that asymmetric blowdown loads resulting from large breaks in main coolant loop piping need not be considered as a design basis.

North Anna Units 1 and 2 were not included with the group of plants for which the unresolved safety issue A-2 was addressed. Therefore to supplement the fracture mecnanics studies performed for the A-2 owner's group a plant specific fracture mechanics study was undertaken for North Anna Units 1 and 2. Westinghouse topical reports WCAP-11163 and 11164 (reference 6, under preparation) will document the results of the plant specific study.

These Westinghouse topical reports WCAP-11163 and 11164, in association with the other references will provide a substantial and adequate basis for limiting postulated design basis flaws in the North Anna Units 1 and 2 stainless steel reactor coolant system piping. The analyses will demonstrate that the probability of rupturing such piping is extremely low under design basis conditions.

The bases for the request for North Anna Units 1 and 2 will be as follows:

1. Extensive operating experience has demonstrated the integrity of the PWR reactor coolant system primary loop including the fact that there has never been a leakage crack.

2. Pre-service and in-service inspections performed on the RCS piping minimize the possibility of flaws existing in such piping. The application of advanced fracture mechanics had demonstrated in other applications that if such flaws exist they will not grow to a leakage crack when subjected to the worst case loading condition over the life of the plant.

3. If a large through-wall flaw is postulated, large margins against unstable crack extension exist for the stainless steel primary coolant piping even if subjected to the safe shutdown earthquake in combination with the loads associated with normal operation.

4. Units have adequate leakage detection systems such that a postulated reference flaw would yield detectable leakage with margin when subjected to normal loading.

This request will also be based upon the improved reliability due to the removal of the snubbers. The removal of snubbers eliminates the potential inadvertent lock-up, bleed rate variances, and hydraulic fluid leakage for these snubbers. The elimination of these snubbers will allow better maintenance of snubbers which remain. The NRC has previously found that the removal of snubbers can enhance reliability as discussed in NUREG-CR-3718 and NUREG-CR-4279. Also, in addition to WCAP-11163 and 11164, which document the plant specific fracture mechanics study, further information will be submitted addressing loading evaluation and leakage detection systems evaluation. These are summarized below.

1. Loading Evaluation:

It is understood that high margins of safety must be retained in the primary component supports. The main reactor coolant loop piping, components and supports evaluation is being performed for the revised support configuration for the loads other than the dynamic effects due to main reactor coolant loop pipe break. This evaluation will be based on the installed and verified reactor coolant loop pipe and equipment support geometry as modified by the removal of these snubbers and by the installation of two rigid restraints per loop. The rigid restraints will be field verified to conform to the analysis after installation. The analysis is being performed by Westinghouse Electric Corporation and Stone and Webster Engineering Corporation with a division of tasks similar to that in the original design analysis.

The snubbers to be eliminated are parallel to the loop piping, and the axial stiffness of the snubbers is small compared to that of the parallel loop piping. Seismic analyses of the primary coolant system with the modified support design will demonstrate that acceptable stress levels are maintained.

This evaluation in addition to demonstrating adequate margins in normal operating and seismic conditions will confirm that maximum forces and moments in the pipe resulting from the elimination of the designated snubbers is still within the applicable forces and moments used in the fracture mechanics study of the reactor coolant loop piping (Reference 6).

2. Leakage Detection Systems Evaluation

The second condition of Generic Letter 84-04 requires that leakage detection systems exist to detect postulated flaws utilizing guidance from Regulatory Guide 1.45, with exception of seismic equipment qualification for the airborne particulate monitor.

The North Anna systems for Reactor Coolant leakage detection are described in UFSAR Section 5.2.4. Conservative calculations of leakage from flaws shown to be stable will be included in the North Anna specific WCAP-11163 and 11164. The equipment provided for leak detection and leak detection operability requirements are delineated in Section 3/4.4.6 of the North Anna Units 1 and 2 Technical Specifications. Because the Technical Specifications require the operability of leak detection systems and because these systems, with margin, are capable of detecting leakage from postulated through-wall flaws, adequate leak detection capability will exist to satisfy the conditions of approval.

References:

1. "Modification of General Design Criterion 4 Requirements for Protection against Dynamic Effects of Postulated Pipe Ruptures," 51 FR 12502, April 11, 1986.

2. WCAP-9558, Revision 2, May 1981, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-wall Crack."

3. WCAP-9787, May 1981, "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation."

4. Letter Report NS-EPR-2519, E. P. Rahe to D. G. Eisenhut, November 10, 1981, Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981.

5. Generic Letter 84-04, February 1, 1984, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."

6. WCAP-11163 Westinghouse Proprietary Class 2 and WCAP-11164 Westinghouse Proprietary Class 3 "Technical Basis for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for North Anna Units 1 and 2."