December 5, 1988

Docket Nos. 50-338
and 50-339

DISTRIBUTION
See attached page

Mr. W. R. Cartwright
Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Cartwright:

SUBJECT: NORTH ANNA UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REACTOR COOLANT PUMP AND STEAM GENERATOR SUPPORTS (TAC NOS. 63577 AND 63578)

The Commission has issued the enclosed Amendment Nos. 107 and 93 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). The amendments add a license condition in response to your letter dated November 6, 1986, as supplemented by letters dated February 24 and March 12, 1987, and March 8 and June 10, 1988.

The amendments permit the redesign of the NA-1&2 reactor coolant pump and steam generator supports since the dynamic effects of postulated primary loop pipe ruptures would be eliminated from the design basis using fracture mechanics "leak-before-break" technology as permitted by the revised General Design Criterion 4 of Appendix A, 10 CFR Part 50.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 107to NPF-4
2. Amendment No. 93to NPF-7
3. Safety Evaluation

cc w/enclosures:
See next page

LA:PDII-2	PM:PDII-2	D:PDII-2	OGC-WF
DMiller	LEngle	HBerkow	OK
11/22/88	11/22/88	11/23/88	12/1/88

DFol
1/1

Mr. W. R. Cartwright
Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

cc:
Mr. William C. Porter, Jr.
County Administrator
Louisa County
P.O. Box 160
Louisa, Virginia 23093

Michael W. Maupin, Esq.
Hunton and Williams
P. O. Box 1535
Richmond, Virginia 23212

Mr. W. T. Lough
Virginia Corporation Commission
Division of Energy Regulation
P. O. Box 1197
Richmond, Virginia 23209

Ellyn R. Weiss, Esq.
Harmon, Weiss and Jordan
2001 S Street NW
Washington, DC 20009

Mr. W. L. Stewart
Senior Vice President - Power
Virginia Electric and Power Co.
Post Office Box 26666
Richmond, Virginia 23261

Mr. Patrick A. O'Hare
Office of the Attorney General
Supreme Court Building
101 North 8th Street
Richmond, Virginia 23219

Resident Inspector/North Anna
c/o U.S. NRC
Senior Resident Inspector
Route 2, Box 78
Mineral, Virginia 23117

C. M. G. Buttery, M.D., M.P.H.
Department of Health
109 Governor Street
Richmond, Virginia 23219

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, Georgia 30323

Mr. G. E. Kane
P. O. Box 402
Mineral, Virginia 23117

Old Dominion Electric Cooperative
c/o Executive Vice President
Innsbrook Corporate Center
4222 Cox Road, Suite 102
Glen Allen, Virginia 23060

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 107
License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated November 6, 1986, as supplemented by letters dated February 24 and March 12, 1987, and March 8 and June 10, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, paragraph 2.F of Facility Operating License NPF-4 is added:

   2.F. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittal dated November 6, 1986 (Serial No. 86-477A).

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Date of Issuance: December 5, 1988

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. NPF-7

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated November 6, 1986, as supplemented by letters dated February 24 and March 12, 1987, and March 8 and June 10, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, paragraph 2.F of Facility Operating License NPF-7 is added:

   2.F. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittal dated November 6, 1986 (Serial No. 86-477A).

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Date of Issuance: December 5, 1988

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 107 AND 93 TO

FACILITY OPERATING LICENSE NO. NPF-4 AND NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated November 6, 1986, and as supplemented by letters dated February 24 and March 12, 1987 and March 8 and June 10, 1988, the Virginia Electric and Power Company (the licensee) proposed amendments to Facility Operating Licenses NPF-4 and NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). The amendments would permit plant operation with the reactor coolant pump and steam generator supports redesigned in accordance with the fracture mechanics "leak-before-break" (LBB) technology as permitted by the revised General Design Criterion 4 (GDC-4) of Appendix A to 10 CFR Part 50. The amendments would add a license condition to Operating Licenses NPF-4 and NPF-7 stating that the design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittal dated November 6, 1986 (Serial No. 86-477A).

The revised GDC-4 is based on the development of advanced fracture mechanics technology using the LBB concept. On October 27, 1987, a final rule was published in the Federal Register (52 FR 41288) to be effective November 27, 1987, amending GDC-4 of Appendix A to 10 CFR Part 50. The revised GDC-4 allows the use of analyses to eliminate from the design basis the dynamic effects of postulated pipe ruptures in high energy piping in nuclear power units. The new technology reflects an engineering advance which allows simultaneously an increase in safety, reduced worker radiation exposures, and lower construction and maintenance costs. Implementation permits the removal of pipe whip restraints and jet impingement barriers as well as other related changes in operating plants, plants under construction, and future plant designs. Containment design and emergency core cooling requirements are not influenced by this modification. The acceptable technical procedures and criteria are defined in NUREG-1061, Volume 3.

Based on the revised GDC-4, the licensee has requested approval for a redesign of the reactor coolant pump and steam generator supports at NA-1&2. The revised GDC-4 eliminates the need for consideration of postulated breaks in the reactor coolant system (RCS) primary loop piping and its effects such as pipe whip, jet impingement, asymmetric pressure loading, and primary component sub-compartment pressurization. Approval of the licensee's request will allow the elimination of certain snubbers which are now required solely to mitigate a pipe rupture event and to replace certain snubbers with rigid restraints which have minimal thermal movement.

The licensee's request is based upon the use of advanced fracture mechanics technology as applied to primary system piping in two Westinghouse Topical Reports: WACAP-9558, Revision 2 (May 1981), "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack"; WACAP-9787 (May 1981), "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation"; and Letter Report NS-EPR-2519, E. P. Rhae (Westinghouse) to D. G. Eisenhut (NRC) dated November 10, 1981. Approval by the NRC of the above Topical Reports is provided in Generic Letter 84-04 dated February 1, 1984, entitled "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."

Generic Letter 88-04 provided the NRC staff Safety Evaluation Report for analysis of materials submitted for a group of utilities operating PWR's to resolve Unresolved Safety 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. NA-1&2 were not included with the group of plants for which the Unresolved Safety Issue A-2 was addressed. Therefore, to supplement the fracture mechanics studies performed for the A-2 Owner's Group, a plant-specific fracture mechanics study was undertaken for NA-1&2. Westinghouse reports WCAP-11163 and -11164, dated August 1986 and entitled "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for North Anna, Units 1 and 2," were submitted by the licensee as part of the amendment request dated November 6, 1986. By letter dated March 8, 1988, the licensee submitted WCAP-11163, Supplement 1, entitled "Additional Information in Support of the Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for North Anna Units 1 and 2," dated January 1988, in response to the staff's request for additional information. The bases for WCAP-11163, Supplement 1 are consistent with the guidelines of NUREG-1061, Volume 3. In addition, the licensee referenced the Westinghouse reports WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems" dated November 1983, and WCAP-10931, "Toughness Criteria for Thermally Aged Cast Stainless Steel," Revision 1, dated July 1986. WCAP-10456 and WCAP-10931 have been previously reviewed by the staff.

As noted above, acceptable technical procedures and criteria are defined in NUREG-1061, Volume 3, and the staff has reviewed and evaluated the licensee's submittal for compliance with the revised GDC-4. A discussion of these matters as well as the staff's findings and evaluation is provided below.

2.0 DISCUSSION

2.1 NA-1&2 Primary Loop Piping

The NA-1&2 primary loop piping consists of 34-inch, 36-inch, and 33-inch nominal diameter hot leg, cross-over leg, and cold leg, respectively. The piping material in the primary loops is cast stainless steel (SA-351 CF8A piping and SA-351 CF8M fittings). The piping is centrifugally cast and the fittings are statically cast. The welding processes used were submerged arc (SAW), shielded metal arc (SMAW), and gas tungsten arc (GTAW). The staff's criteria for evaluation of compliance with the revised GDC-4 are discussed in Chapter 5.0 of Reference 7 and are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight, and thermal expansion) due to normal operation, and the forces and moments associated with the safe shutdown earthquake (SSE). These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments, and safe ends.

(2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue, or water hammer are not likely, should be provided. Relevant operating history should be cited, which includes system operational procedures; system or component modification; water chemistry parameters, limits, and controls; and resistance of material to various forms of stress corrosion and performance under cyclic loadings.

(3) The materials data provided should include types of materials and materials specifications used for base metal, weldments, and safe ends; the materials properties including the fracture mechanics parameter "J-integral" (J) resistance (J-R) curve used in the analyses; and long-term effects such as thermal aging and other limitations to valid data (e.g., J maximum, and maximum crack growth).

(4) A through-wall flaw should be postulated at the highest stressed locations determined from criterion (1) above. The size of the flaw should be large enough so that the leakage is assured of detection with at least a factor of 10 using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

(5) It should be demonstrated that the postulated leakage flaw is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be at least 1.4 and should be determined by a flaw stability analysis, i.e., that the leakage-size flaw will not experience unstable crack growth even if larger loads (larger than design loads) are applied. However, the final rule permits a reduction of the margin of 1.4 to 1.0 if the individual normal and seismic (pressure, deadweight, thermal expansion, SSE, and seismic anchor motion) loads are summed absolutely. This analysis should demonstrate that crack growth is stable and the final flaw size is limited, such that a double-ended pipe break will not occur.

(6) The flaw size should be determined by comparing the leakage-size flaw to the critical-size flaw. Under normal plus SSE loads, it should be demonstrated that there is a margin of at least 2 between the leakage-size flaw and the critical-size flaw to account for the uncertainties inherent in the analyses and leakage detection capability. A limit-load analysis may suffice for this purpose; however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.

The staff has evaluated the information presented in WCAP-11163 and WCAP-11163, Supplement 1 for compliance with the revised GDC-4. Furthermore, the staff performed independent flaw stability computations using an elastic-plastic fracture mechanics procedure developed by the staff in NUREG/CR-4572, "NRC Leak-Before-Break (LBB, NRC) Analysis Method for Circumferentially Through-Wall Cracked Pipes Under Axial Plus Bending Loads," May 1986.

On the basis of its review, the staff finds the NA-1&2 primary loop piping in compliance with the revised GDC-4. The following paragraphs in this section present the staff's findings.

(1) Normal operating loads, including pressure, deadweight, and thermal expansion, were used to determine leak rate and leakage-size flaws. The flaw stability analyses performed to assess margins against pipe rupture at postulated faulted load conditions were based on normal plus SSE loads. In the stability analysis, the individual normal load components were summed algebraically and the seismic loads were then added absolutely. In the leak rate analysis, the individual normal load components were summed algebraically. Leak-before-break evaluations were performed for the limiting location in the piping.

(2) For Westinghouse facilities, there is no history of cracking failure in RCS primary loop piping. The RCS primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history totals over 450 reactor-years, including 5 plants each having over 16 years of operation and 15 other plants each with over 11 years of operation.

(3) The material tensile and fracture toughness properties were provided in WCAP-11163 and WCAP-11163, Supplement 1. Because there are cast stainless steel piping (and fitting) and associated welds in the NA-1&2 primary loop, the thermal aging toughness properties of cast stainless steel materials were estimated according to procedures in WCAP-10456 and WCAP-10931. The material tensile properties were estimated using generic procedures. For flaw stability evaluations, the lower-bound stress-strain properties were used. For leakage rate evaluations, the average stress-strain properties were used.

(4) NA-1&2 have RCS pressure boundary leak detection systems which are consistent with the guidelines of Regulatory Guide 1.45 such that a leakage of one gallon per minute (gpm) can be detected. The calculated leak rate through the postulated flaw is large relative to the staff's required sensitivity of the plant's leak detection systems; the margin is at least a factor of 10 on leakage and is consistent with the guidelines of NUREG-1061, Volume 3.

(5) In the flaw stability analyses, the margin in terms of load for the leakage-size flaw under normal plus SSE loads exceeds 1.4 and is consistent with the guidelines of NUREG-1061, Volume 3.

(6) The margin between the leakage-size flaw and the critical size flaw was also evaluated in the flaw stability analyses. The margin in terms of flaw size exceeds 2 and is consistent with the guidelines of NUREG-1061, Volume 3.

Based on the above, the staff concludes that the NA-1&2 primary loop piping complies with the revised GDC-4 according to the criteria in NUREG-1061, Volume 3.

2.2 NA-1&2 Reactor Coolant Pump and Steam Generator Supports Redesign

The supporting system redesign would permit NA-1&2 to reduce the number of large bore hydraulic snubbers at the RCP and SG supporting systems of the reactor coolant loops (RCLs). The new design will eliminate two snubbers from every RCP supporting system, four snubbers from every SG supporting system, and replace two snubbers with rigid struts in every SG supporting system; thereby reducing the number of snubbers in each RCL in NA-1&2 by eight. The technical basis for this redesign is the use of "leak-before-break" analyses, which were used to justify elimination of the dynamic effects of postulated pipe ruptures from the design of primary piping systems. The analyses performed show that the redesign will be able to withstand all remaining loadings, including those caused by the SSE and the limiting high energy line breaks at branch nozzles. Specifically, the analytical results indicated that stresses in the RCL piping are within the UFSAR allowables with adequate margins of safety.

NA-1&2 each have three RCLs in their reactor coolant systems. Each RCL has one RCP and one SG. Identical designs are used to support the RCPs and likewise for the SGs. The RCP support is a frame structure with two snubbers installed to restrict movement parallel to the cold leg. The SGs are supported laterally at two levels. The upper SG support consists of four snubbers tangentially arranged around a ring girder. The lower SG support is a rectangular-cubic structure interconnected with the RCP support by four snubbers to restrict movement perpendicular to the hot leg direction. It is also mounted with two other snubbers to restrict movement parallel to the hot leg.

The proposed redesign would eliminate the two snubbers parallel to the hot leg from the lower SG support, two snubbers in the RCP support, and two snubbers from the SG-RCP support interconnection. Further, it will replace the two snubbers in the hot leg direction by two rigid struts in the upper RCP support.

Westinghouse Reports WCAP-11163 and 11164 provide the basis for the redesign with reduced loading level, which in turn, requires less support rigidity during the remaining dynamic events required for the plant's design. Loadings considered in the redesign are those cause by deadweight, internal pressure, thermal movement, seismic events which include the Operating Bases Earthquake (OBE) and SSE, and postulated pipe ruptures at nozzles.

Two independent analyses were performed to verify the adequacy of the redesign. One was performed by Westinghouse using the WESTDYN computer code to obtain RCL equipment and piping stresses. The other one was performed by the Stone & Webster Engineering Corporation (SWEC) using the STARDYN computer code to obtain support loads. Both the WESTDYN and the STARDYN computer codes were approved by NRC in 1974. Both mathematical models used mass and stiffness representations to simulate one complete RCL. Three dimensional seismic analyses were performed by using peak broadening amplified response spectra with equipment damping values of half and one percent for OBE and SSE respectively. The three directional seismic responses were combined by the square-root-sum-of-squares (SRSS) method. Combination of closely spaced modes was conducted according to the 10% method recommended by Regulatory Guide 1.92, Rev. 1. Good agreement was found between the Westinghouse and the SWEC results.

The redundancy of equipment or components in a system should be taken into account when the reliability of equipment or components is considered. If a failure occurs in a redundant structure, the consequences may not be as serious as in a structure with less inherent redundancy. In the case of the RCL with many snubbers removed, the remaining snubbers will need to be more reliable since the level of redundancy has been reduced. The licensee plans to replace the remaining snubber units with hydraulic snubbers manufactured by Taylor Devices. The service record of these snubbers has shown no service-oriented failures were ever discovered. Since NA-1&2 has committed to the maintenance practices recommended by the manufacturers, high reliability of snubber performance will be assured. The staff finds the licensee has proposed redesigning the NA-1&2 SG and RCP supporting systems by applying approved technology and by using qualified equipment.

3.0 EVALUATION

The staff has reviewed the information submitted by the licensee and has performed independent flaw stability computations. On the basis of its review, the staff concludes that the NA-1&2 primary loop piping complies with the revised GDC-4 according to the criteria in NUREG-1061, Volume 3. Thus, the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of NA-1&2 is sufficiently low such that dynamic effects associated with postulated pipe breaks need not be a design basis. In addition, the NA-1&2 proposed redesign at the SG and RCP supporting systems uses approved technology and using qualified equipment. Therefore, the staff finds the results of the supporting analyses to be acceptable. Based on all of the above, the staff finds the proposed redesign of the NA-1&2 RCP and SG supports to be acceptable and in conformance with GDC-4 of Appendix A, 10 CFR Part 50.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 5, 1988

Principal Contributors:

S. Lee
H. K. Shaw
L. B. Engle

DATED: December 5, 1988

AMENDMENT NO. 107 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1
AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
L. Engle
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
T. Meek (8), P1-137
Wanda Jones, P-130A
E. Butcher, 11/F/23
ACRS (10)
GPA/PA
ARM/LFMB
B. Wilson, R-II
S. Lee
H. Shaw

cc: Plant Service list