VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

July 6, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 726D
GLD/RWC:brh:0555C
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

SUPPLEMENT TO AMENDMENT TO OPERATING LICENSES NPF-4 AND NPF-7
NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
REACTOR COOLANT SYSTEM TEMPERATURE OF 587.8°F

In our letter dated December 30, 1982 (Serial No. 726), Vepco requested an amendment to Operating Licenses NPF-4 and NPF-7 to allow operation of North Anna Unit Nos. 1 and 2 at a reactor coolant system average temperature of 587.8°F. This letter provides in Attachment 1 supplemental information in answer to questions discussed with members of the Staff's Reactor Systems Branch on April 27, 1983 and June 21, 1983.

Should you have any further questions, please contact us at your earliest convenience.

Very truly yours,

W. L. Stewart

W. L. Stewart

Attachment

(1) Response to Reactor Systems Branch for North Anna Operation at RCS Average Temperature of 587.8°F

cc: Mr. James P. O'Reilly
Regional Administrator
Region II

Mr. Charlie Price
Department of Health
109 Governor Street
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Mr. Richard Barrett
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Mr. M. B. Shymlock
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North Anna Power Station

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ATTACHMENT 1

RESPONSE TO REACTOR SYSTEMS BRANCH

FOR NORTH ANNA OPERATION

AT RCS AVERAGE TEMPERATURE OF 587.8^{o}F

ATTACHMENT 1

RESPONSE TO NRC REACTOR SYSTEMS BRANCH QUESTIONS
ON NORTH ANNA 7.5°F UPRATING SUBMITTAL NSSS TRANSIENT ANALYSES

Q1. Loss of Normal Feedwater (FSAR p. 15.2-50, UFSAR p. 15.2-40)

Assumption 4 in the existing FSAR states that the following was used in the calculation:

> "A heat transfer coefficient in the steam generator associated with reactor coolant system natural circulation."

Why did the uprating submittal delete this assumption?

A1. The original FSAR analysis assumed a loss of offsite power occurred simultaneously with the loss of normal feedwater, so the SG heat transfer coefficient used was one appropriate for natural circulation conditions. A later reanalysis in response to an NRC question assumed offsite power was available. The 7.5°F uprating analysis employed current methodology, which is to analyze the loss of feedwater with and without offsite power available. In each case, the SG heat transfer coefficient is calculated as a function of local SG conditions.

Q2. Loss of Normal Feedwater (FSAR p. 15.2-52, UFSAR p. 15.2-41)

The FSAR states that at no time during the transient does the water level in the steam generators receiving auxiliary feedwater uncover the tubesheet and there is no water relief from the pressurizer. There are figures of pressurizer and steam generator water level response in the FSAR, but no steam generator water level figure in the uprating submittal. What happens to SG water level during the transient?

A2. The loss of normal feedwater accident analyzed for the FSAR was simulated using the BLKOUT Code (Ref. WCAP-7501). The digital program computes pertinent variables including the SG level, pressurizer water level, and reactor coolant average temperature; however, for the 7.5°F uprating analysis, the LOFTRAN code (Ref. WCAP-7378, Rev. 3) was used to simulate the loss of normal feedwater accident. The LOFT4 digital program computes pertinent variables including the steam generator mass versus time, pressurizer water volume, and reactor coolant average temperature. The capacity of the auxiliary feedwater pumps are such that the water level in the fed steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the reactor coolant system relief or safety valves.

Q3. Excessive Load Increase (FSAR Fig. 15.2-33, UFSAR Fig. 15.2-46)

The existing UFSAR results (Fig. 15.2-46) for an excessive load increase at BOL without automatic rod control show the nuclear power increasing from 1.02 to approximately 1.08 times nominal. Explain the difference between this and the uprating analysis, in which the power increases from 1.02 to 1.05 times nominal.

A3. The UFSAR analysis was performed by changing only the moderator density coefficient in the BOL and EOL cases. Current methodology analyzes the excessive load increase event for minimum and maximum reactivity feedback conditions. This requires minimum feedback values for the Doppler temperature coefficient, the Doppler power defect, beta and lambda star; in addition to the moderator density coefficient. Using these minimum reactivity feedback coefficients yields a lower steady state power level than that predicted in the UFSAR.

Attached are revised FSAR Figures 15.2-33, -34, -35, -36, -37, -38, -39, and -40, and Table 15.2-1 which have been relabeled to reflect the minimum and maximum feedback conditions included in the 7.5^{o}F reanalyses.

Q4. Accidental Depressurization of the Main Steam System

The uprating submittal includes neither a reanalysis of this event, nor a statement explaining why it was not reanalyzed.

A4. In FSAR Section 15.2.13, the existing analysis of this event assumes a steam release equal to the maximum capacity of any single steam dump or safety valve. The transient is assumed to be initiated at no load plant conditions, which are unchanged by the proposed 7.5^{o}F increase in RCS average temperature. It is typically analyzed as a special case of the main steamline break accident, which is listed as one of the zero power transients requiring no reanalysis at the uprated conditions (enclosure 1, Page 14 of 7.5^{o}F uprating submittal).

Q5. Main Feedline Rupture (proposed Fig. 15.4.2-7A, 7B)

Why are the results of the uprating analysis (Figs. 15.4.2-7A, 7B), e.g., pressurizer pressure and core average temperature, less severe than the existing FSAR results (UFSAR Figs. 15.4-30, 31)?

A5. The change in the North Anna feedwater system from a headered (which was assumed in the FSAR analysis) to a one-on-one arrangement (which was assumed in the 7.5^{o}F uprating analysis) is the major reason for the reduced severity of the main feedline rupture results.

Also, the results presented in FSAR are calculated using the MARVEL code, whereas the results presented in the 7.5^{o}F uprating analysis are calculated by the LOFTRAN code, using assumptions consistent with the Westinghouse Feedline Break topical WCAP-9230.