VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

February 3, 1997

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 96-409A
NO&LS/ETS
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
SUPPLEMENTAL INFORMATION CONCERNING CORE DESIGN MARGINS AND FUEL MANAGEMENT FOR USE OF FCF LEAD TEST ASSEMBLIES

This letter provides the supplemental information which was discussed in teleconference calls between Virginia Electric and Power Company and the NRC staff. This information is provided to support the NRC staff's review of the license amendment and exemption request to permit the use of lead test assemblies in North Anna Units 1 and 2. The license amendment and exemption request were submitted in our letter Serial No. 96-409, dated September 4, 1996. The discussions with the NRC staff provided data concerning margins which are inherent in the proposed core design containing the lead test assemblies. Specifically, on November 19, 1996 and December 19, 1996, staff from our Nuclear Analysis and Fuel Department provided numerical values of calculated margin in two key power distribution parameters, Enthalpy Rise Hot Channel Factor ($F\Delta H$) and Heat Flux Hot Channel Factor (FQ). Attachment 1 provides more complete calculated results which have been obtained from analysis of the proposed North Anna 1, Cycle 13 core design containing the lead test assemblies. Information pertaining to fuel management considerations for the cores which will contain the lead test assemblies is also provided.

In light of recent control rod insertion problems at some units, which may be related to fuel assembly guide thimble conditions, the NRC requested the Company consider RCCA control rod drag testing on the lead test assemblies. The Company has concluded that this additional testing would contribute useful data which is fully consistent with the overall program objectives. Therefore, Virginia Electric and Power Company will perform drag testing on the FCF lead test assemblies as described in Attachment 1. Although it is intended to perform the testing as proposed, the scope and schedule of such testing may be modified if necessary to conform to overall refueling outage constraints.

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Also, during the telephone conference calls with the NRC Staff, alternative words were discussed for the proposed changes to Technical Specifications Section 5.3.1. Attachment 2 provides revised words for the proposed change to Section 5.3.1 that more closely conform to the Standard Technical Specifications. The proposed changes in Attachment 2 supersede the respective changes contained in the initial September 4, 1996 submittal.

The proposed revision to Section 5.3.1 of the Technical Specifications does not affect the original basis for our determination that the changes do not involve a significant hazards consideration. The proposed Technical Specifications changes have been reviewed by the Station Nuclear Safety and Operating Committees and the Management Safety Review Committee .

The commitments made in this letter are summarized in Attachment 3. If you have any further questions or concerns, please contact us.

Very truly yours,

J.P. O'Hanlon

J. P. O'Hanlon
Senior Vice President - Nuclear

Attachments

cc: U. S. Nuclear Regulatory Commission
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Attachment 1

Fuel Management and Inherent Core Design Margins Applicable to Use of FCF Lead Test Assemblies in North Anna 1 Cycle 13

General Description

The FCF lead test assemblies are scheduled to be first irradiated in North Anna 1, Cycle 13. As described in the Company's license amendment request (Reference 1), the lead test assemblies are functionally equivalent to the resident Westinghouse fuel and incorporate several advanced features. The intent of the program is to provide meaningful performance data regarding these features by irradiating the lead test assemblies in a manner which is representative of typical reload fuel. The description below provides additional discussion of fuel management plans, core design margins and inspection and testing plans for the lead test assemblies.

Fuel Management Plans

The Company intends to irradiate the lead test assemblies in a manner typical of that employed in the low leakage cores designed for the North Anna reactors. This fuel management scheme typically involves irradiation in a relatively high power location during the first cycle; placement in a similar interior core location for the second cycle (sometimes under an RCCA); and use of the fuel near or on the periphery of the core in the third cycle, which provides a combination of shielding and most efficient use of the remaining fissionable material. The specific plans for use of the lead test assemblies in accordance with this approach are provided below.

In the first cycle of irradiation (Cycle 13), the proposed core loading pattern incorporates the lead test assemblies in the positions indicated in Figure 1 with 'LTA C13.' In these locations, the lead test assemblies will experience moderately severe duty typical of normal reload fuel. To ensure that the existing safety analyses based on the resident Westinghouse fuel remain applicable, calculation results for the proposed Cycle 13 loading pattern (see Tables 1a, 1b, 2) have confirmed that the FCF assemblies are not in the highest fuel rod power density locations in the core. In addition, the lead test assemblies do not have the highest cycle-averaged assembly average power. The lead test assemblies are not limiting in any key core operation parameters related to rod power density.

This proposed loading pattern and its calculated results are subject to change in the event that a redesign is necessary for Cycle 13 operation. The Company will inform NRC if this should occur so that the requirements for additional reporting of data for the redesigned core can be established.

Figure 1 indicates by 'LTA C14A' and 'LTA C14B' some of the locations under consideration for placement of the lead test assemblies during the second cycle of irradiation. These locations are typical positions for such once-burned assemblies in North Anna core designs. It should be noted that locations denoted with 'LTA C14B' are under RCCAs. Placement under RCCAs (which is typical for once-burned assemblies) is being considered for the lead test assemblies. A final decision concerning the second cycle position for the lead test assemblies will be made during North Anna 1, Cycle 13 operation. This decision will consider both the advantage of additional incore RCCA experience with this advanced fuel assembly design (beyond the data obtained from control rod drag testing discussed later) and the potential risks of such use in

rodded core locations. In their third cycle, the lead test assemblies are most likely to be placed on the peripheral core locations denoted with 'LTA C15' on Figure 1.

This description has been provided to indicate the proposed core management plans for use of the FCF lead test assemblies. These plans are intended to provide the most meaningful data from the lead test assembly program, but are subject to change if required by overall core management strategies.

<u>Core Design Margins</u>

In teleconferences between NRC and Company staff on November 19 and December 19, 1996, calculated results from the North Anna 1, Cycle 13 core design were discussed. The results for two key core design parameters were presented: Enthalpy Rise Hot Channel Factor (FΔH) and Heat Flux Hot Channel Factor (FQ). These discussions summarized the calculated values of minimum margins available between the peak FΔH and FQ for the core and the FΔH and FQ Technical specification limits. Tables 1a, 1b and 2 provide more complete calculated results which have been obtained from analysis of the Cycle 13 design containing the lead test assemblies.

Table 1a provides calculated values and relevant margins for FΔH as a function of burnup throughout Cycle 13, assuming the previous cycle (Cycle 12) shuts down at a burnup of 16100 MWD/MTU (referred to as low window). Data are presented for the peak calculated FΔH of the Westinghouse fuel and the lead test assemblies. Percentage margin is presented for comparison between: 1) Westinghouse fuel and the Technical Specifications limit; and 2) lead test assemblies and the Technical Specifications limit. Table 1b presents the same data assuming that Cycle 12 operates to a burnup of 17100 MWD/MTU (high window). The following summary is obtained from Tables 1a and 1b and corresponds to the values discussed during the November 19, 1996 teleconference:

Westinghouse fuel minimum margin to FΔH limit = 2.5% at 11000 MWD/MTU
LTA minimum margin to FΔH limit = 7.7% at 15000 MWD/MTU

Table 2 presents calculated parameter values and relevant margins for FQ. These data were calculated with the approved Relaxed Power Distribution Control methodology (Reference 2) which involves calculations at three burnup values. Results are presented for both the low end and high end of the of Cycle 12 burnup window. Data are presented for the peak calculated FQ of the Westinghouse fuel and the lead test assemblies. Percentage margin is presented for comparison between: 1) Westinghouse fuel and the Technical Specifications limit; and 2) lead test assemblies and the Technical Specifications limit. The following summary of the Table 2 data corresponds to the values discussed during the December 19, 1996 teleconference:

Westinghouse fuel minimum margin to FQ limit (at the peak FQ location) =
6.5% at MOC burnup - High Window
LTA minimum margin to FQ limit (at the peak FQ location) =
10.6% at EOC burnup - High Window

The data in Tables 1a, 1b and 2 are calculated for Cycle 13 operation. Calculated margins for future cycles in which the lead test assemblies are used will differ from the data presented for

Cycle 13. As indicated in both teleconferences, $F_{\Delta H}$ and FQ margin to the limit is generally greater in the second and third cycles of irradiation.

LTA Inspection and Testing

During fabrication, the lead test assemblies were characterized to determine baseline values of dimensions and other features for later comparison with post-irradiation examination results. This characterization included, but was not necessarily limited to: certification of the composition and material properties of the fuel rod cladding and guide thimble materials; dimensional characterization of the fuel rods, including cladding diameters and wall thicknesses, plus the lengths of all peripheral rods; dimensional characterization of the guide thimbles, including lengths and diameters; dimensional characterization of the fuel assemblies, including assembly lengths, fuel rod to top nozzle gaps, grid envelopes, grid elevations, and water channel measurements; and measurement of the force (rotation torque) required for the top nozzle quick disconnect mechanism.

Post irradiation examinations of the lead test assemblies will be performed during the program as permitted by the North Anna refueling schedule. In addition to full length visual examinations, we currently anticipate these examinations will include: measurement of fuel assembly length and bow, holddown spring compression testing, functional testing of the quick disconnect locking mechanism, oxide measurements on fuel rods and guide thimbles, measurements of fuel rod diameter, and shoulder gap measurements (fuel rod length determination).

The North Anna lead test assembly program has been structured from the outset to provide the most meaningful irradiation experience and relevant data with which to characterize performance of the advanced fuel design features. This approach has resulted in a significant amount of analysis and evaluation to validate the design for use in the North Anna cores. In light of recent control rod insertion problems at some units which may be related to fuel assembly guide thimble conditions, the NRC staff discussed our plans to perform RCCA control rod drag testing on the lead test assemblies to provide additional performance data for this advanced fuel design. Virginia Electric and Power Company has concluded that this additional testing would contribute useful data which is fully consistent with the overall program objectives. Therefore, Virginia Electric and Power Company will perform drag testing on the FCF lead test assemblies. The proposed approach would involve testing analogous to that which we have performed for the resident Westinghouse fuel assemblies during recent refueling outages. While it is intended to obtain as much data as is reasonable, this testing (and other planned post-irradiation lead test assembly testing) will be subject to refueling outage schedule constraints. The current test plan involves performing control rod drag tests on all four lead test assemblies at the end of each cycle of irradiation. An initial drag test of the lead test assemblies may also be performed prior to irradiation to establish a benchmark result for comparison of post-irradiation test results.

References

1. Letter, James P. O'Hanlon to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company; North Anna Power Station Units No. 1 and 2 - Notification of Intention to Use Lead Fuel Assemblies with Advanced Cladding Materials," Serial No. 96-409, September 4, 1996.

2. K. L. Basehore, et al., "Virginia Power Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," Topical Report VEP-NE-1-A, March 1986.

Table 1a

Calculated Values of Normal Operation Enthalpy Rise Hot Channel Factor ($F\Delta H$)[1]
For North Anna 1 Cycle 13 Core Design Containing FCF Lead Test Assemblies
Low Window End of Cycle 12 Burnup (16100 MWD/MTU)

N1C13 Burnup (MWD/MTU)	W Fuel Peak $F\Delta H$	LTA Peak $F\Delta H$	Margin(%)[2,3] (W to Limit)	Margin(%)[2,3] (LTA to Limit)
0	1.359	1.305	5.3	9.1
150	1.343	1.298	6.4	9.5
1000	1.326	1.300	7.6	9.4
3000	1.350	1.304	5.9	9.1
5000	1.366	1.304	4.8	9.1
7000	1.381	1.305	3.8	9.1
9000	1.393	1.306	2.9	9.0
11000	1.394	1.308	2.9	8.8
13000	1.383	1.313	3.6	8.5
15000	1.370	1.321	4.5	7.9
17000	1.356	1.322	5.5	7.9
18600 (EOR)	1.343	1.319	6.4	8.1

[1] Unrodded Hot Full Power Results - LTAs will exhibit more margin with control rods at the rod insertion limits
[2] Limit value for $F\Delta H$ is 1.435
[3] Margin (%) to the limit is calculated as follows:

$$\text{Margin (\%)} = \frac{[\text{Value - Limit}]}{\text{Limit}} \times 100$$

Table 1b

Calculated Values of Normal Operation Enthalpy Rise Hot Channel Factor ($F\Delta H$)[1]
For North Anna 1 Cycle 13 Core Design Containing FCF Lead Test Assemblies
High Window End of Cycle 12 Burnup (17100 MWD/MTU)

N1C13 Burnup (MWD/MTU)	W Fuel Peak $F\Delta H$	LTA Peak $F\Delta H$	Margin(%)[2,3] (W to Limit)	Margin(%)[2,3] (LTA to Limit)
0	1.349	1.314	6.0	8.4
150	1.337	1.302	6.8	9.3
1000	1.337	1.303	6.8	9.2
3000	1.353	1.306	5.7	9.0
5000	1.370	1.306	4.5	9.0
7000	1.386	1.307	3.4	8.9
9000	1.398	1.308	2.6	8.8
11000	1.399	1.310	2.5	8.7
13000	1.389	1.317	3.2	8.2
15000	1.376	1.324	4.1	7.7
17000	1.362	1.325	5.1	7.7
18100 (EOR)	1.353	1.324	5.7	7.7

[1] Unrodded Hot Full Power Results - LTAs will exhibit more margin with control rods at the rod insertion limits

[2] Limit value for $F\Delta H$ is 1.435

[3] Margin (%) to the limit is calculated as follows:

$$\text{Margin (\%)} = \frac{[\text{Value - Limit}]}{\text{Limit}} \times 100$$

Table 2

Calculated Values of Heat Flux Hot Channel Factor (FQ)
For North Anna 1 Cycle 13 Core Design Containing FCF Lead Test Assemblies
(Values Presented are Normal Operation FQ)[1]

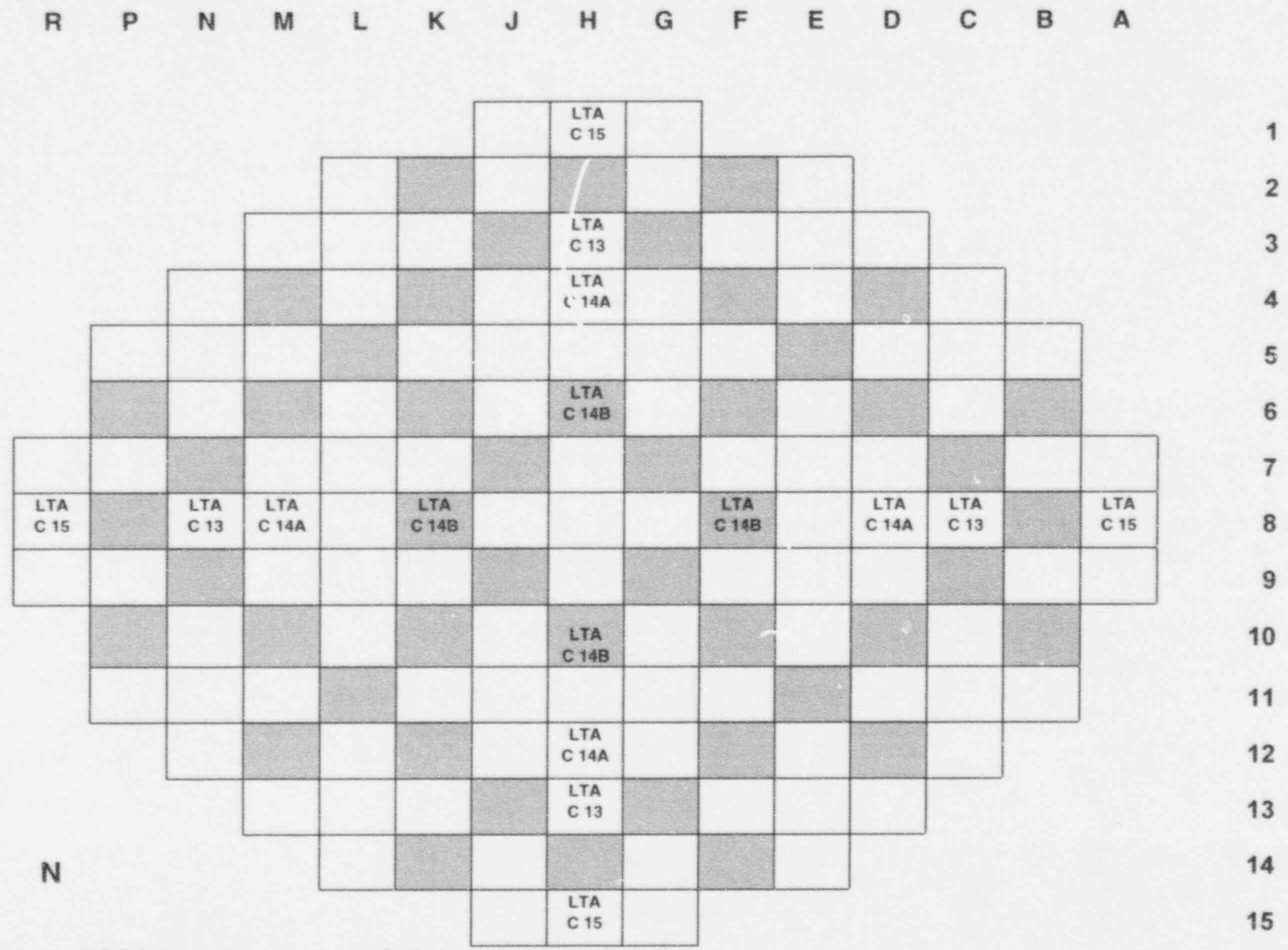
N1C13 Burnup (Time in Cycle-Window)	W Fuel Peak FQ	LTA Peak FQ	Margin(%)[2,3] (W to Limit)	Margin(%)[2,3] (LTA to Limit)
BOC - Low	1.9156	1.8645	12.5	14.9
MOC - Low	2.0406	< 1.9434	6.8	> 11.3
EOC - Low	1.9787	1.9494	9.6	11.0
BOC - High	1.9461	1.8680	11.1	14.7
MOC - High	2.0466	< 1.9491	6.5	> 11.0
EOC - High	1.9975	1.9573	8.8	10.6

[1] Includes all Relaxed Power Distribution Control power shapes within assumed ΔI operational limits
[2] Limit value for FQ is 2.19
[3] Margin (%) to the limit is calculated at the point of peak FQ as follows:

$$\text{Margin (\%)} = \frac{[\text{Value - Limit}]}{\text{Limit}} \times 100$$

NORTH ANNA UNIT 1

Figure 1



Note: RCCA Locations are shaded