VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 9, 1987

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

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

Serial No. 87-337
E&C/NAS:vlh
Docket No. 50-280
License No. DPR-32

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT NO. 1
CORE INTEGRITY EVALUATION FOR RECENT FLOW REDUCTION EVENT

On May 16, 1987, Surry Unit 1 experienced an automatic shutdown on low reactor coolant flow in one loop. It was subsequently determined that the cause of the trip was the mechanical failure of a reactor coolant loop stop valve, resulting in the partial insertion of the valve disk into the flow path of the affected loop.

On May 27, 1987, a telephone conference was held between Virginia Electric and Power Company and the NRC (Region II and NRR Staff) to discuss the event, and in particular to provide an assessment of the possibility of a violation of the Departure from Nucleate Boiling (DNB) design criterion as a result of the event.

The attachment summarizes the evaluation methodology and results. As discussed with the Staff, we have concluded that no violation of any core design limits resulted from the event, and in fact the transient was bounded by those previously assessed in the UFSAR against the acceptance criteria for ANS Condition II (moderate frequency) events.

Please contact us at your convenience if you have any additional questions.

Very truly yours,

W. L. Stewart

Attachment

1. Core Integrity Evaluation

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cc: U. S. Nuclear Regulatory Commission
Region II
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Mr. W. E. Holland
NRC Senior Resident Inspector
Surry Power Station

Mr. Chandu P. Patel
NRC Surry Project Manager
PWR Project Directorate No. 2
Division of PWR Licensing-A

ATTACHMENT

Core Integrity Evaluation

Evaluation of Possibility of Core DNB
Surry 1 Dropped Stop Valve Disk Event

On May 16, 1987, Surry Unit 1 experienced an automatic shutdown on low reactor coolant flow in one loop. It was subsequently determined that the cause of the trip was the mechanical failure of a reactor coolant loop stop valve, resulting in the partial insertion of the valve disk into the flow path of the affected loop.

On May 27, 1987, a telephone conference was held between Virginia Electric and Power Company and the NRC (Region II and NRR Staff) to discuss the event, and in particular to provide an assessment of the possibility of a violation of the Departure from Nucleate Boiling (DNB) design criterion as a result of the event.

As discussed at that time, we have evaluated the effects of the event from a core DNB standpoint and have concluded, based on a conservative analysis, that the DNB design basis for moderate frequency anticipated transients (i.e. ANS Condition II events) was met throughout the event. The basis for this conclusion is as follows:

a) The observed flow rate in the faulted loop at the end of the event was 45% of full flow. Conservatively neglecting the increase in flow in the unaffected loops that occurs when a loop coasts down, the core flow at the end of the event was $(1.0 + 1.0 + 0.45)/3 = 0.817$ times full flow.

b) Any transient effect of the reactor trip on low loop flow on reducing the core power heat and flux has been conservatively ignored.

c) The most recent measured loop flows were, on the average, 6% higher than the thermal design flow rate assumed in the safety analysis. This flow margin was ignored.

d) A COBRA IIIC/MIT (Ref. 1) statepoint calculation was performed with current licensed production models based on the following conservative assumptions -

* Core flow = 81.7% of thermal design value

* Core power and heat flux = hot full power + maximum calorimetric calibration error

* Core exit pressure = nominal controlled pressurizer pressure less deadband and uncertainties. (Core pressures are higher than the pressurizer pressure - this is a DNB benefit which is ignored)

* Core inlet temperature = nominal hot full power value plus maximum instrument uncertainties and control deadband error.

* Hot channel radial peaking factor (F-delta-H) assumed at hot full power Tech. Spec. limit = 1.55

The most recent measured value of F-delta-H (including uncertainty) prior to the event was about 6% below the limit. Control bank D was essentially fully withdrawn (227 steps) at the time of the event, so the measured F-delta-H is considered representative of actual core conditions preceeding the event.

The results of the analysis showed a minimum DNBR of 1.461 in the hot channel. This compares with the DNB design limit of 1.30. It should also be noted that the currently applicable complete loss of flow results presented in the UFSAR yield a minimum DNBR which is slightly lower than 1.461 when analyzed with the same COBRA model. This indicates that the limiting core conditions realized during the event were less severe than those predicted in the existing docketed loss of flow analysis.

Based on the results of this evaluation, we conclude that there was no violation of the DNB design criterion for moderate frequency events, fuel integrity was not compromised, and the transient experienced by Surry Unit 1 was no more severe than the loss of flow events currently considered in the UFSAR.

References

1. Sliz, F. W. and Basehore, K. L. "VEPCO Reactor Core Thermal - Hydraulic Analysis using the COBRA IIIC/MIT Computer Code", Vep-FRD-33-A, October, 1983.