VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 24, 1994

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No. 94-337
SPS/BCB-CGL R5'
Docket No. 50-280
License No. DPR-32

Gentlemen:

**VIRGINIA ELECTRIC AND POWER COMPANY**
**SURRY POWER STATION UNIT 1**
**NRC INSPECTION REPORT NO 50-280/94-11**
**REPLY TO A NOTICE OF VIOLATION**

Your letter dated May 26, 1994, provided the results of NRC Inspection Report No. 50-280/94-11. The report identified a violation for failure to revise the Unit 1 steam flow calorimetric computer program to incorporate the changes implemented by a revision to an engineering calculation. As a result of this event, we have completed a thorough root cause evaluation and will be implementing the necessary corrective actions to prevent recurrence.

To further strengthen our programs, we plan to perform assessments of the change control and reactivity management processes. In addition, we will verify through a sampling process that other scaling revisions are appropriately implemented.

The response to the notice of violation is attached. Please contact us if you have any questions or require additional information.

Very truly yours,

J. P. O'Hanlon

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

Attachment



IE01
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cc: U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N.W.
Atlanta, Georgia 30323

Mr. M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

# REPLY TO A NOTICE OF VIOLATION
# NRC INSPECTION CONDUCTED April 3 - May 7, 1994
# SURRY POWER STATION UNIT 1
# INSPECTION REPORT NO. 50-280/94-11

## NRC COMMENT:

During an NRC Inspection conducted on April 3 through May 7, 1994, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the violation is listed below:

> 10 CFR 50, Appendix B Criterion III, Design Control as implemented by section 17.2.3 of the Operational Quality Assurance Program Topical Report, VEP-1-5A (Updated) require that measures be established to assure that design requirements be correctly translated into specifications, drawings, procedures, and instructions.
>
> Electrical Engineering Implementing Procedure EE-029, Calculation Controlling Procedure, revision 2, stated scope (2.2) requires that calculation results be effectively communicated to the applicable power station.
>
> Engineering Calculation No. EE-0418, Determination of Feedwater Flow and Steam Flow Transmitter's Calibration Spans from CHEMTRAC and Flowcalc Data Resulting from Special Test 1-ST-300, revision 1, determined revised main steam flow transmitter span and instrument scale values.
>
> Contrary to the above, design requirements of Calculation EE-0418, revision 1, were not correctly translated into instructions, in that, the revised main steam flow span and scale values were not incorporated into the steam flow calorimetric computer program. As a result, the power range nuclear instruments were improperly adjusted and on March 30, 1994, the unit operated for one shift at an average power level of 2453 megawatts thermal which exceeded the licensed maximum power level of 2441 megawatts thermal.
>
> This is a Severity Level IV Violation (Supplement I).

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**SURRY POWER STATION UNIT 1**
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1) **Reason for the Violation, or if Contested, the Basis for Disputing the Violation**

The reason for this violation was a programmatic weakness in the change control process for implementation of engineering calculations affecting station instrumentation. Adequate controls were not in place to ensure that the Prodac 250 (P-250) computer system FLOWCALC program, which computes main steam and feedwater flows, was changed to reflect the implementation of a revision to Engineering Calculation No. EE-0418. Engineering Calculation No. EE-0418 is entitled "Determination of Surry Power Station, Unit 1, Feedwater Flow and Steam Flow Transmitter Calibration Spans from Chemtrac and Flowcalc Data Resulting from Special Test Procedure 1-ST-300."

Revision 1 to Engineering Calculation No. EE-0418 was issued in late 1992 to correct a minor conservative error that had been identified in the results of Special Test 1-ST-300, "Calibration of Steam Flow Transmitters Using Chemtrac Chemical Tracer Method." The Document Management Information System (DMIS) was updated to reflect this revision. In addition, memoranda were sent to the Station Engineering Department and the P-250 computer programmer to notify both groups of the revision.

As part of the Procedures Upgrade Program, "Steam Line Flow Protection [Loop] Channel Calibration" Instrument Periodic Test procedures were revised in late 1993 using Engineering Calculation No. EE-0418, Revision 1. These procedures, which are performed on a refueling frequency, were used to span and scale the steam flow transmitters during the 1994 Unit 1 refueling outage. The FLOWCALC program was not revised at that time due to a lack of clearly defined responsibility for coordinating and verifying implementation of instrumentation-related changes (i.e., documents, equipment, and software). Therefore, at the time the unit was returned to service, the steam flow instrumentation was calibrated based on Engineering Calculation No. EE-0418, Revision 1, whereas the FLOWCALC program was based on Revision 0 of the calculation.

Output from the FLOWCALC program is used by the steam flow/feedwater flow calorimetric program (CALCALC) to provide an indication of reactor power based on steam flow or feedwater flow. Due to the error in the FLOWCALC program, the indicated reactor power was approximately one percent less than actual reactor power. As a result, the unit operated for one shift at an average power level of approximately 2453 megawatts-thermal, 100.5%. This condition exceeded the steady state power level authorized by the facility operating license.

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1) **Reason for the Violation, or if Contested, the Basis for Disputing the Violation (Continued)**

Engineering evaluated the effect of the FLOWCALC program error on the values in the CALCALC program and performed conservative calculations to determine the actual reactor power level during the event. These calculations indicated that reactor power did not exceed 101.78%. The power level experienced is bounded by Surry's accident analyses, which includes 2.0% uncertainties.

2) **Corrective Steps Which Have Been Taken and the Results Achieved**

When the unit reached 100% power as indicated by the steam flow calorimetric, control room operators noted that the main generator gross output indication was higher than expected. Station management reviewed the condition and directed operations personnel to reduce power to 820 megawatts-electrical. Unit power was reduced to approximately 98% as indicated by the steam flow calorimetric.

On March 31, 1994, engineering personnel began an investigation to determine the cause of the high electrical output indication from the main generator. The investigation determined that the FLOWCALC program had not been updated to incorporate the most recent revision of the main steam flow transmitter span values. To immediately correct the effect of this error on reactor power indication, the power range nuclear instrumentation was adjusted and the reactor coolant system delta-T indications were calibrated based on the feedwater flow calorimetric. These indications had previously been based on the steam flow calorimetric.

A Root Cause Evaluation (RCE) team was promptly formed to investigate the cause of the event and to recommend corrective actions. The results of the investigation were presented to management. In addition, the RCE and its recommendations were reviewed by the NRC resident inspectors, as discussed in NRC Inspection Report No. 50-280/94-11.

The FLOWCALC program was revised to incorporate the correct steam flow transmitter pressure span values.

## 2) Corrective Steps Which Have Been Taken and the Results Achieved (Continued)

The power range nuclear instrumentation was adjusted to the corrected steam flow calorimetric and the unit was returned to 100% power on April 5, 1994.

## 3) Corrective Steps That Will be Taken to Avoid Further Violations

Modifications to the instrumentation change control process (including change implementation) are planned to ensure that setpoint/scaling changes to station instrumentation are effectively controlled. The specifics of these modifications are currently under development. As an interim measure, for scaling packages being issued, a coordinated review will be conducted to ensure effective control of associated changes prior to implementation.

Actions to address related RCE recommendations are being evaluated by management for implementation.

Development of 1) modifications to the instrumentation change control process and 2) completion of the evaluation of/development of a plan to implement the related RCE recommendations is scheduled for completion by September 1, 1994. The implementation of the plan will be monitored by management through the Level I process and the status will be periodically communicated to the NRC resident inspectors.

Operations and engineering personnel and technical staff managers will receive training on this event and the interrelationship and limitations of the various reactor power indications.

## 4) The Date when Full Compliance Will be Achieved

Compliance was achieved on March 31, 1994, when the FLOWCALC program was revised to reflect the implementation of Engineering Calculation No. EE-0418, Revision 1.