VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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

Serial No. 94-416
NL&P/CGL R4'
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PLANNED ACTIONS AND IMPLEMENTATION SCHEDULE
ASSOCIATED WITH VIOLATION 94-11-01
AND RELATED ROOT CAUSE EVALUATION

NRC Inspection Report 50-280, -281/94-11 identified a violation (VIO 94-11-01) related to the failure to revise the Surry Unit 1 steam flow calorimetric computer program to incorporate changes implemented by an engineering calculation prior to unit restart following the refueling outage. Our June 24, 1994 response (letter Serial No. 94-337) identified actions planned to address these circumstances and indicated that recommendations from the related Root Cause Evaluation (RCE) were under evaluation for implementation. By a July 5, 1994 letter, the NRC requested a more detailed explanation of our planned corrective actions and implementation schedule. The requested information is provided in the attachment. Although the attachment specifically addresses implementation at Surry, it should be noted that comparable corrective actions will also be implemented at North Anna, as applicable.

Very truly yours,

Robert F Saunders for

James P. O'Hanlon
Senior Vice President - Nuclear

Attachment - Planned Actions and Implementation Schedule Associated with Violation 94-11-01 and Related Root Cause Evaluation

AE01

cc: U. S. Nuclear Regulatory Commission
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Mr. M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

Mr. R. D. McWhorter
NRC Senior Resident Inspector
North Anna Power Station

ATTACHMENT

PLANNED ACTIONS AND IMPLEMENTATION SCHEDULE ASSOCIATED WITH VIOLATION 94-11-01 AND RELATED ROOT CAUSE EVALUATION

Our response to NRC Inspection Report 50-280, -281/94-11 was provided by a June 24, 1994 letter (Serial No. 94-337). That letter identified planned actions, as well as indicated that the recommendations from the related Root Cause Evaluation (RCE) were being evaluated for implementation. Evaluation of the RCE recommendations has been completed. The following information provides more detailed information regarding our planned activities, including the approved RCE recommendations, and the planned implementation schedules.

Modifications to the Instrumentation Change Control Process

A new program is being developed to address scaling control. The program requirements will ensure effective control of scaling changes to station instrumentation. The new program will include:

- Definition of responsibilities for the scaling change process,
- Identification of station procedures, documents, and software affected by proposed scaling changes,
- Designated reviews of proposed scaling changes which could impact reactivity or equipment that affects nuclear safety, and
- Update of priority procedures, documents, and software prior to implementation of proposed scaling changes.

Implementation of the new program is scheduled for November 11, 1994.

Training of Personnel and Managers on this Event

Training of operations and engineering personnel and technical staff managers regarding the Surry Unit 1 operation above 100% event, as well as the interrelationship and limitations of the various reactor power indications, has been initiated. This training will be completed by December 31, 1994 as part of the Licensed Operator Requalification Program and the Technical Staff Continuing Training. (It should be noted that these training programs normally include topics addressing industry/plant operational events and experiences, such as this event.)

Assessment of the Change Control Processes

An assessment of the change control processes used has been initiated. The purpose of this assessment is to perform a general review of the change control processes and methods for initiating, evaluating, and implementing change. The processes will be

defined and evaluated. Particular attention will be given to those processes which are identified as problematic or inadequate. This assessment is scheduled for completion by October 31, 1994.

Assessment of the Reactivity Management Controls

An assessment of the reactivity management controls was performed to ensure that the Reactor Engineers are cognizant of any changes that could affect reactivity or reactor power indications. With respect to reactivity management controls, the Reactor Engineer's awareness of hardware changes, procedure changes, instrumentation changes, software changes, and routine plant operations was evaluated. The conclusions reached and planned program enhancements are discussed in the following paragraphs.

Regarding hardware changes, the existing design change process was determined to be adequate to ensure that Reactor Engineering is cognizant of any change that could affect reactivity. Therefore, no enhancement to the design change process is required.

Evaluation of the procedure change process concluded that new and revised procedures that could potentially affect reactivity may not always be reviewed by Reactor Engineering. Therefore, a revision to the procedure change process is planned to ensure that a reactivity management review is performed, where necessary.

With respect to instrumentation changes, the new program to ensure effective control of scaling changes to station instrumentation (discussed above) will assure the appropriate involvement of Reactor Engineering.

Related to software control, there are P-250 programs that are integral to the reactivity management program, such as the reactor power calorimetric program. It was determined that changes to these computer programs should have a Reactor Engineering review to assess any impact the changes may have on the reactivity management program prior to implementation. Therefore, the administrative procedure addressing computer software control will be revised accordingly.

Finally, Reactor Engineering's cognizance of routine plant operations was also assessed. It was determined that Reactor Engineering routinely uses various means (including daily planning meetings held at both stations, review of Operations Morning Reports and Shift Orders, review of plant computer data and reports, and daily visits to the control room) to maintain cognizance of planned operations. However, these practices will be standardized to enhance communication and involvement in this area.

The revisions to the procedure change process, the computer software control administrative procedure, and practices by Reactor Engineering to maintain cognizance of plant operations, which are discussed above, are planned for implementation prior to the restart of Surry Unit 2 following the next refueling outage (i.e., April 1995, based on the current outage schedule).

Verification of Appropriate Implementation of Other Scaling Revisions

A review was performed on completed scaling packages and calculations that have the potential to require that other loops or components be adjusted. The review included documents prepared or revised from January 1, 1991 to the present. It has been concluded that there are no other loops or components that are required to be changed as a result of the scaling packages and calculations reviewed.

Evaluation of the Need to Provide Alternate Means of Verifying Reactor Power Prior to Exceeding 96% Power

The current practice during return to full power is to perform a verification of reactor power by evaluating several alternate indications. These alternate indications include nuclear instrumentation (NIs), delta temperature (ΔTs), steam generator pressure, first stage turbine impulse pressure, megawatts electric, and feed flow/steam flow calorimetrics. This verification is performed after each refueling outage during power escalation at 30%, 70%, and 100% power. In this evaluation, any acceptance criteria that may be placed on these indications are not formally documented and are based on experience and previous historical data.

Based on the review of the current practice, it was concluded that enhancements were required to ensure the accuracy of the calorimetric program. First, the computer code performing the calculation will be verified prior to startup following a refueling or other extended outage, where required. The verification will include reviewing key parameters to ensure plant modifications which affect input parameters are linked to appropriate software changes. Secondly, it was concluded that the current practice of evaluating alternate power indications could be enhanced by documenting the methodology to be employed and the acceptance criteria for each indicator to the extent practical. Values deviating from the acceptance criteria would require investigation and resolution prior to continued power escalation. The enhanced reactor power verification will be performed prior to exceeding 96% power following refueling outages and extended outages where modifications have been made that could affect these indications.

These enhancements to current practice will strengthen the verification of reactor power and are planned for implementation prior to the restart of Surry Unit 2 following the next refueling outage (i.e., April 1995, based on the current outage schedule).

Evaluation of the Need to Incorporate Oversight of Calculations and Calculation Implementation into the Normal Activities of Quality Assurance (QA) - Engineering

Based on the review conducted, it was determined that QA oversight of calculations and analysis is appropriate. This oversight function has been incorporated into the QA Critical Attribute Matrix (CAM) for Design Control. The CAMs are tools that define critical attributes for QA audit, inspection, and/or assessment for specified functional areas.