VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

August 5, 1993

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

Serial No.	93-396
NL&P/CGL	R0"
Docket Nos.	50-338
	50-339
	50-280
	50-281
License Nos.	NPF-4
	NPF-7
	DPR-32
	DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
SURRY POWER STATION UNITS 1 AND 2
RESPONSE TO GENERIC LETTER 93-04
ROD CONTROL SYSTEM FAILURE AND WITHDRAWAL
OF CONTROL ROD ASSEMBLIES, 10CFR50.54 (f)

On June 21, 1993, the NRC issued Generic Letter 93-04, entitled "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10CFR50.54(f)." The generic letter notifies addressees about a potential single failure vulnerability within the Westinghouse solid state rod control system that could cause an inadvertent withdrawal of control rods in a sequence resulting in a power distribution not considered in the design basis analyses. The generic letter also requires that affected addressees provide the NRC with information describing their plant-specific findings related to this issue and actions taken.

Specifically, Generic Letter 93-04 requires that, within 45 days from the date of the generic letter, affected addressees provide an assessment of whether or not the licensing basis for each facility is still satisfied with regard to the requirements for system response to a single failure in the rod control system (GDC 25 or equivalent) (Required Response 1(a)). If the assessment in Required Response 1(a) indicates that the licensing basis is not satisfied, then the licensee must provide an assessment of the impact of potential single failures in the rod control system on the licensing basis of the facility (Required Response 1(b), part one), describe compensatory short-term actions (Required Response 1(b), part two), and within 90 days provide a plan and schedule for long-term resolution (Required Response 2).

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Subsequent correspondence between the Westinghouse Owners Group and the NRC resulted in schedular relief for Required Responses 1(a) and 1(b), part one (July 26, 1993 NRC letter to Mr. Roger Newton of the Westinghouse Owners Group). Since, we are participating in the Westinghouse Owners Group initiatives addressing this issue, the Required Responses 1(a) and 1(b), part one, will be included with our 90 day response.

Generic Letter 93-04 Required Response 1(b), part two, for North Anna and Surry is provided in the attachment. This response summarizes the compensatory actions taken by Virginia Electric and Power Company in response to the Salem rod control system failure event. This response also provides a summary of the results of the generic safety analysis program conducted by the Westinghouse Owners Group and its applicability to North Anna and Surry. This transmittal completes our 45 day required response to Generic Letter 93-04 (as amended by July 26, 1993 NRC letter to Mr. Roger Newton of the Westinghouse Owners Group).

As discussed in the attachment, the rod control system design, short-term compensatory actions, as well as the current testing, training, and operating practices, at both stations provide a high degree of assurance that an unchecked rod withdrawal will not occur. Additionally, the generic analysis performed by the Westinghouse Owners Group and the plant-specific applications demonstrate that the design DNBR limits are met with margin for the entire spectrum of asymmetric rod withdrawal events analyzed for North Anna and Surry.

Should you have questions regarding this information or require additional information, please contact us.

Very truly yours,

W. L. Stewart
Senior Vice President - Nuclear

Attachment - Response to Generic Letter 93-04 for North Anna and Surry Power Stations - Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10CFR50.54(f)

cc: U. S. Nuclear Regulatory Commission
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by W. L. Stewart who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 5 day of Aug, 1993.

My Commission Expires: Aug 31, 1994.

Carolyn Moore
Notary Public

(SEAL)

ATTACHMENT

RESPONSE TO GENERIC LETTER 93-04 FOR NORTH ANNA AND SURRY POWER STATIONS - ROD CONTROL SYSTEM FAILURE AND WITHDRAWAL OF ROD CONTROL CLUSTER ASSEMBLIES, 10CFR50.54(f)

Compensatory Actions

The purpose of this discussion is to provide a response to the three areas of compensatory short-term actions identified by the NRC (Required Response 1(b), part two) and any additional compensatory actions judged to be appropriate.

1. **Additional cautions or modifications to surveillance and preventive maintenance procedures.**

   Response

   The Westinghouse Owners Group (WOG) and Westinghouse have concluded that an increased frequency in surveillance testing is not required or appropriate in response to the Salem rod control system failure event. Increased surveillance testing is contrary to the general trend and philosophy of surveillance testing relaxation in that increased testing can, in and of itself, result in higher rates of system and component failures.

   Virginia Electric and Power Company is in agreement with the WOG and Westinghouse conclusion that an increased frequency in surveillance testing is not required, based primarily on our current practices.

   At North Anna, rod movement surveillance tests are performed monthly to verify rod operability. The rod deviation alarm is verified to be operable weekly. The frequency of this surveillance ensures that rod misalignments of greater than 12 steps can be detected in a reliable manner. The rod deviation alarm is required to be operable during rod movement testing.

   At Surry, rod movement surveillance tests are performed biweekly to verify rod operability. Surveillance test procedure revisions have been made to verify that the computer printout annunciator for the rod control system incorrect sequencing and the rod bank to limit alarms are lighted during testing, provided the P-250 computer is available. If the P-250 computer is not available or if a rod is misaligned, individual rod positions are monitored and logged on an increased frequency. In addition, a new procedure was developed to test alarms monthly for rod position deviations of 9 and 21 steps.

2. **Additional administrative controls for plant startup and power operation.**

Both the WOG and Westinghouse have concluded that startup by dilution is not required in response to the Salem rod control system failure event. In actual operation, an operator would be aware of abnormal rod movement and terminate rod demand prior to reaching criticality.

Virginia Electric and Power Company is in agreement with the WOG and Westinghouse conclusion that startup by dilution is not required or desired. Plant startup and power operation procedures for North Anna and Surry are consistent with the operational philosophy discussed in the previous paragraph. Specifically, during approaches to criticality, rod control is performed manually. Therefore, it is highly unlikely that during startup an unchecked rod withdrawal resulting in criticality would occur.

In addition, a review of North Anna and Surry station deviation reports indicates that any deviations (2 to 3 steps) in rod group or bank alignment have resulted in the operators taking actions specified by procedure, as well as initiating the appropriate notifications and troubleshooting.

3. **Additional instructions and training to heighten operator awareness of potential rod control system failures and to guide operator response in the event of a rod control system malfunction.**

Both the WOG and Westinghouse have recommended that licensees provide additional discussion, training, standing orders, etc. to ensure that their operators are aware of what occurred at Salem. The recommendations of the Westinghouse Nuclear Safety Advisory Letter, which was subsequently endorsed by the July 2, 1993 WOG letter (OG-93-42), recognize the benefits of ensuring that plant operators are knowledgeable of the Salem rod control system failure event.

At both North Anna and Surry, operations personnel were informed of the Salem event and alerted to be particularly sensitive to rod position indication during either automatic or manual control rod manipulations. This was accomplished by memorandum, required reading, and/or shift briefings.

In addition, control rod misalignments and rod control system failures are an integral part of both operator initial and requalification training at both stations.

Summary of the Generic Safety Analysis Program

Introduction

As part of the Westinghouse Owners Group initiative, the WOG Analysis subcommittee has developed a generic approach to demonstrate that for Westinghouse plants there is no safety significance for an asymmetric Rod Cluster Control Assembly (RCCA) withdrawal. The purpose of the program is to analyze a series of asymmetric rod withdrawal cases from both subcritical and power conditions to demonstrate that departure from nucleate boiling (DNB) does not occur.

The current Westinghouse analysis methodology for the bank withdrawal at power and from subcritical uses point-kinetics and one-dimensional kinetics transient models, respectively. These models use conservative constant reactivity feedback assumptions, which result in an overly conservative prediction of the core response for these events.

A three-dimensional spatial kinetics/systems transient code (LOFT5/SPNOVA) is used to show that the localized power peaking is not as severe as current codes predict. The 3-D transient analysis approach uses a representative standard 4-loop Westinghouse plant with conservative reactivity assumptions. Limiting asymmetric rod withdrawal statepoints (i.e., conditions associated with the limiting time in the transient) are established for the representative plant that can be applied to all Westinghouse plants. Differences in plant designs are addressed by using conservative adjustment factors to make a plant-specific DNB assessment.

Description of Asymmetric Rod Withdrawal

The accidental withdrawal of one or more RCCAs from the core is assumed to occur, which results in an increase in the core power level and the reactor coolant temperature and pressure. If the reactivity worth of the withdrawal rods is sufficient, the reactor power and/or temperature may increase to the point that the transient is automatically terminated by a reactor trip on a high nuclear flux or Over-Temperature Delta-T (OTDT) protection signal. If the reactivity rise is small, the reactor power will reach a peak value and then decrease due to the negative feedback effect caused by the moderator temperature rise. The accidental withdrawal of a bank or banks of RCCAs in the normal overlap mode is a transient that is specifically considered in plant safety analysis reports. The consequences of a bank withdrawal accident meet Condition II criteria (no DNB). If, however, it is assumed that less than a full group or bank of control rods is withdrawn and these rods are not symmetrically located around the core, this can cause a "tilt" in the core radial power distribution. The "tilt" could result in a radial power distribution peaking factor that is more severe than is normally considered in the plant safety analysis report and, therefore, cause a loss of DNB margin. Due to the imperfect mixing of the fluid exiting the core before it enters the hot legs of the reactor coolant loops, there can be an imbalance in the loop temperatures and, therefore, in the measured values of T-avg and delta-T, which are used in the Over-Temperature Delta-T protection system for the core. The radial power "tilt" may

also affect the ex-core detector signals used for the high nuclear flux trip. The axial offset (AO) in the region of the core where the rods are withdrawn may become more positive than the remainder of the core, which can result in an additional DNB penalty.

Methods

The LOFT5 computer code is used to calculate the plant transient response to an asymmetric rod withdrawal. The LOFT5 code is a combination of an advanced version of the LOFT4 code (Reference 1), which has been used for many years by Westinghouse in the analysis of the RCS behavior to plant transients and accidents, and the advanced nodal code SPNOVA (Reference 2).

LOFT5 uses a full core model, consisting of 193 fuel assemblies with one node per assembly radially and 20 axial nodes. Several "hot" rods are specified with different input multipliers on the hot rod powers to simulate the effect of plants with different initial $F\Delta H$ values. A "hot" rod represents the fuel rod with the highest $F\Delta H$ in the assembly, which is calculated by SPNOVA within LOFT5. DNBRs are calculated for each hot rod within LOFT5 with a simplified DNB-evaluation model using the WRB-1 correlation. The DNBRs resulting from the LOFT5 calculations are used for comparison purposes.

A more detailed DNBR analysis is done at the limiting transient statepoints from LOFT5 using THINC-IV (Reference 3) and the Revised Thermal Design Procedure (RTDP). RTDP, which maximizes DNBR margins, applies to Westinghouse plants and has been approved by the NRC. The LOFT5-calculated DNBRs are conservatively low when compared to the THINC-IV results.

Assumptions

The initial power levels chosen for the performance of bank and multiple RCCA withdrawal cases are 100%, 60%, 10% and hot zero power (HZP). These power levels are the same powers considered in the RCCA Bank Withdrawal at Power and Bank Withdrawal from Subcritical events presented in the plant Safety Analysis Reports. The plant, in accordance with RTDP, is assumed to be operating at nominal conditions for each power level examined. Therefore, uncertainties will not affect the results of the LOFT5 transient analyses. For the at-power cases, all reactor coolant pumps are assumed to be in operation. For the HZP case (subcritical event), only 2 of 4 reactor coolant pumps are assumed to be in operation. A "poor mixing" assumption is used for the reactor vessel inlet and outlet mixing model.

Results

A review of the results presented in Reference 4 indicates that for the asymmetric rod withdrawal cases analyzed with the LOFT5 code, the DNB design basis is met. As demonstrated by the A-Factor approach (described below) for addressing various combinations of asymmetric rod withdrawals, the single most-limiting case is

plant-specific and is a function of rod insertion limits, rod control pattern, and core design. The results of the A-factor approach also demonstrate that the cases analyzed with the LOFT5 computer code are sufficiently conservative for a wide range of plant configurations and for various asymmetric rod withdrawals. In addition, when the design FΔH is taken into account on the representative plant, the DNBR criterion is met for the at-power cases. At HZP, a worst-case scenario (3 rods withdrawn from three different banks, which is not considered to be possible) shows a non-limiting DNBR. These results are applicable to Westinghouse plants.

Plant Applicability

The 3-D transient analysis approach uses a representative standard 4-loop Westinghouse plant with bounding reactivity assumptions with respect to the core design. This results in conservative asymmetric rod(s) withdrawal statepoints for the various asymmetric rod withdrawals analyzed. The majority of the cases analyzed either did not generate a reactor trip or were terminated by a high neutron flux reactor trip. For the OTDT reactor trip, no credit is assumed for the f(ΔI) penalty function. The f(ΔI) penalty function reduces the OTDT setpoint for highly skewed positive or negative axial power shapes. Compared to the plant-specific OTDT setpoints including credit for the f(ΔI) penalty function, the setpoint used in the LOFT5 analyses is conservative (i.e., for those cases that tripped on OTDT). A plant-specific OTDT setpoint with the f(ΔI) penalty function will result in an earlier reactor trip than the LOFT5 setpoint. This ensures that the statepoints generated for those cases that trip on OTDT are conservative for all Westinghouse plants.

With respect to the neutronic analyses, an adjustment factor ("A-factor") was calculated for a wide range of plant types and rod control configurations. The A-factor is defined as the ratio between the design FΔH and the change in the maximum transient FΔH for the symmetric and asymmetric RCCA withdrawal cases. An appropriate and conservative plant-specific A-factor was calculated and used to determine the corresponding DNBR penalty or benefit. With respect to the thermal-hydraulic analyses, differences in plant conditions (including power level, RCS temperature, pressure, and flow) are addressed by sensitivities performed using THINC-IV. These sensitivities are used to determine additional DNBR penalties or benefits. Uncertainties in the initial conditions are accounted for in the DNB design limit. Once the differences in plant design were accounted for by the adjustment approach, plant-specific DNBR calculations can be generated for Westinghouse plants.

Conclusions

Using this approach, the generic analysis and the plant-specific applications demonstrate that for North Anna and Surry DNB does not occur for the worst-case asymmetric rod withdrawal.

Specifically, Virginia Electric and Power Company has reviewed the WOG assumptions and methods, including initial conditions, accident modeling, and application of DNBR sensitivities to plant-specific parameter variations between the

generic analysis and North Anna Units 1 and 2 and Surry Units 1 and 2. We concur with the WOG conclusion that the design DNBR limits are met with margin for the entire spectrum of asymmetric rod withdrawal events analyzed for North Anna and Surry.

References

1. Burnett, T. W. T., et al, "LOFTRAN Code Description," WCAP-7907-A, April 1984.

2. Chao, Y. A., et al, "SPNOVA - A Multi-Dimensional Static and Transient Computer Program for PWR Core Analysis," WCAP-12394, September 1989.

3. Friedland, A. J. and S. Ray, "Improved THINC-IV Modeling for PWR Core Design," WCAP-12330-P, August 1989.

4. Huegel, D. et al, "Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal," WCAP-13803, August 1993.