TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

August 5, 1993

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328
		50-390
		50-391

SEQUOYAH NUCLEAR PLANT (SQN) AND WATTS BAR NUCLEAR PLANT (WBN) - TRANSMITTAL OF RESPONSE TO GENERIC LETTER (GL) 93-04 ROD CONTROL SYSTEM FAILURE AND WITHDRAWAL OF ROD CLUSTER ASSEMBLIES

Pursuant to the requirements of 10 CFR 50.54(f), on Monday, June 21, 1993, the NRC issued GL 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies," to all licensees with the Westinghouse Rod Control System (except Haddam Neck) for action.

The generic letter requires that, within 45 days from the date of the generic letter, each addressee 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). If the assessment (Required Response 1.(a)) indicates that the licensing basis is not satisfied, then the licensee must describe compensatory short-term actions consistent with the guidelines contained in the generic letter, and within 90 days, provide a plan and schedule for long-term resolution (Required Response 1.(b)). Subsequent correspondence between the Westinghouse Owners Group and the NRC resulted in schedular relief for Required Response 1.(a) (NRC Letter to Roger Newton dated July 26, 1993). This portion of the required actions will now be included with the 90-day licensee response.

06-107

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cc (Enclosures):

Mr. D. E. LaBarge, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy Daisy, Tennessee 37379

NRC Resident Inspector
Watts Bar Nuclear Plant
P.O. Box 700
Spring City, Tennessee 37381

Mr. Mark Proviano
Westinghouse Electric Corporation
P.O. Box 355, ECE 4-08
Pittsburgh, Pennsylvania 15230-0355

Mr. P. S. Tam, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ENCLOSURE 1

RESPONSE TO NRC GENERIC LETTER 93-04

Compensatory Actions

The purpose of this discussion is to provide a response to the three areas of compensatory short-term actions for SQN identified by the NRC (require sponse 1.(b)) and any additional compensatory actions judged to be appropriate.

1. **"Additional caution or modifications to surveillance and preventive maintenance procedures" -**

   Westinghouse (W) did not make any initial recommendations regarding surveillance or preventative maintenance procedures. Based on the response provided in the July 2, 1993, Westinghouse Owners Group's (WOG's) OG-93-42, Generic Assessment of the Plant-Specific Compensatory Actions Regarding Salem Rod Control System Event, there was no need to increase the frequency of testing on a permanent or generic basis. Public Service Electric and Gas Company (PSE&G) had committed to a temporary increase in testing, but only until it was demonstrated that the rod control system was operating properly and with confidence. A recommendation was made by WOG for utilities to ensure that their surveillance testing will demonstrate rod control system operability and address maintenance trouble-shooting. 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. Therefore, the WOG and W have concluded that increased frequencies in surveillance testing is not required in response to the Salem rod control system failure event.

   TVA concurs with the WOG and W that increased testing is not required at this time. The present practice at SQN is to exercise the control rods at least ten steps every 31 days for Modes 1 and 2. Additionally, when the reactor trip breakers are closed, the control rods are exercised at least ten steps every 31 days for Modes 3, 4, and 5. Additional checks performed after a refueling outage are the rod drop timing test, the functional check of the rod control logic cabinet, and the rod position indication instrumentation calibration. SQN has reviewed the operational history and the system has had very few operational problems. Based upon this, SQN does not intend to increase the surveillance frequency.

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

PSE&G committed the Salem units to startup by dilution. As stated in OG-93-42, neither W nor the WOG has endorsed this requirement. In actual operation, the operators would be aware of abnormal rod movement and terminate rod demand prior to ever reaching criticality. The operator would be manually controlling the rod withdrawal such that the detection of rod mis-stepping in under one minute would be reasonable. In fact, as demonstrated during the R. E. Ginna event, abnormal rod motion was terminated after only one step both in automatic and manual rod control. It is entirely too unrealistic to believe that the operators would permit an unchecked rod withdrawal during startup such that criticality would be reached. Thus, the WOG and W have concluded that startup by dilution is not required in response to the Salem rod control system failure event.

TVA concurs with the WOG and W that dilution to criticality for every startup is not required. Due to different core parameters at different times in life, pulling rods may be the more expeditious means of startup (rods in manual control), any rod deviation would be immediately addressed. However, SQN does dilute to criticality for the initial startup after a refueling outage. TVA's present start-up procedures contain adequate precautions and warnings.

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

Recommendation 4 presented in the W NASL 93-07, is being implemented. Operator awareness has been provided by issuing the NSAL to all licensed operators as a training letter. As part of the licensed operator's requalification program, rod control system malfunctions are addressed, and the plant specific procedure for rod control system failures is included in the licensed operator's required reading list.

ENCLOSURE 2

SUMMARY OF THE GENERIC SAFETY ANALYSIS PROGRAM

Introduction

As part of the Westinghouse Owners Group (WOG), the WOG Analysis subcommittee is working on a generic approach to demonstrate that for all Westinghouse (W) plants there is no safety significance for an asymmetric 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 DNB does not occur.

The current W 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 being 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 W 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 which can be applied to all W 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 withdrawn 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 which 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 which 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 W 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 assemblies with one node per assembly radially and 20 axial nodes. Several "hot" rod are specified with different input multipliers on the hod 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 $F_{\Delta H}$ values in the assembly, and 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 applies to all W plants, maximizes DNBR margins, is approved by the NRC, and is licensed for a number of W plants. 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 were 100 percent, 60 percent, 10 percent, 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 hot zero power case (subcritical event), only 2/4 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 demonstrates that the cases analyzed with the LOFT5 computer code are sufficiently conservative for a wide range of plant configurations for various asymmetric rod withdrawals. In addition, when the design **$F_{\Delta H}$** 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 different banks which is not possible) shows a non-limiting DNBR. This result is applicable to all other W plants.

Plant Applicability

The 3-D transient analysis approach uses a representative standard 4 Loop W 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 rector trip or were terminated by a High Neutron Flux reactor trip. For the Overtemperature Delta-T reactor trip, no credit is assumed for the **$f(\Delta I)$** 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 **$f(\Delta I)$** 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 **$f(\Delta I)$** penalty function will resulting 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 W 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 $F_{\Delta H}$ from 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 all W plants.

Conclusion

Using this approach, the generic analyses and their plant-specific application demonstrate that for SQN DNB does not occur for their worst-case asymmetric rod withdrawal.

TVA hereby submits its response (Enclosure 1) to the GL as it applies to SQN. This response summarizes the compensatory actions taken by TVA in Required Response 1.(b) to the Salem rod control system failure event. No short-term compensatory actions are required for WBN, at this time, since WBN is under construction. Enclosure 2 provides a summary of the results of the generic safety analysis program conducted by the Westinghouse Owners Group and its applicability to SQN. No commitments have been made in this letter. TVA considers this action to be complete with respect to the 45 day required response to GL 93-04 (as amended by July 26 NRC letter to Roger Newton).

If you have any questions, please telephone me at (615) 751-2687.

Sincerely,

Bruce S. Schofield
Manager
Nuclear Licensing and Regulatory Affairs

Sworn to and subscribed before me
this 5th day of August 1993

Deborah J. Smith
Notary Public

My Commission Expires 10-6-93

Enclosures
cc: See Page 3

REFERENCES

1. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-79-A, April 1984.

2. Chao, Y.A., et al., "SPNOVA - 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.