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August 5, 1993

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
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ULNRC-2832

Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT
ROD CONTROL SYSTEM FAILURE AND WITHDRAWAL OF
ROD CONTROL CLUSTER ASSEMBLIES, 10 CFR 50.54(f)
Reference: NRC Generic Letter 93-04, dated June 21, 1993

Pursuant to the requirements of 10 CFR 50.54(f), the NRC issued Generic Letter 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies," on June 21, 1993 and was addressed to licensees with the Westinghouse Rod Control System for action and to all other licensees for information.

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. 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 Mr. Roger Newton dated July 26, 1993). This portion of the required response will now be included with the 90 day licensee response.

Union Electric Co. hereby submits its 45 day response to the Generic Letter as it applies to Callaway Plant. This response summarizes the compensatory actions

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taken by Union Electric in response to the Salem rod control system failure event. It also provides a summary of the results of the generic safety analysis program conducted by the Westinghouse Owners Group and its applicability to Callaway Plant. Union Electric considers this action to be complete with respect to the 45 day required response to Generic Letter 93-04 (as amended by July 26 NRC letter to Mr. Roger Newton). The 90 day response will be submitted by September 20, 1993.

If you have any questions concerning this matter, please contact me.

Very truly yours,

Al Passwater

for Donald F. Schnell

JMC/dls

Attachments

STATE OF MISSOURI)
) S S
CITY OF ST. LOUIS)

Alan C. Passwater, of lawful age, being first duly sworn upon oath says that he is Manager, Licensing and Fuels (Nuclear) for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Alan C Passwater
Alan C. Passwater
Manager, Licensing and Fuels
Nuclear

SUBSCRIBED and sworn to before me this 5th day of August, 1993.

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NOTARY PUBLIC — STATE OF MISSOURI
MY COMMISSION EXPIRES APRIL 22, 1997
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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)) and any additional compensatory actions judged to be appropriate.

1. "additional cautions or modifications to surveillance and preventive maintenance procedures" -

   Westinghouse did not make any initial recommendations regarding surveillance or preventative maintenance procedures. Based on the response provided in OG-93-42, there was no perceived need to increase the frequency of testing on a permanent or generic basis 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 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 Westinghouse have concluded that increased frequencies in surveillance testing is not required in response to the Salem rod control system failure event.

   Callaway Plant surveillance procedure for Control Rod Partial Movement is used to verify the operability of each control and shutdown rod pursuant to surveillance requirement 4.1.3.1.2 of the Technical Specifications. Precautions in the procedure state, "To verify proper movement, monitor both Digital Rod Position Indication (DRPI) and Bank Step Counters for the rods being tested." and "Do not allow any rod to deviate from its group demand counter indication more than $\pm$ 12 steps.".

   These precautions give clear guidance to acceptable operation of the Rod Control System during surveillance testing.

   Post-maintenance testing utilizes the same procedure and precautions to assure proper rod movement.

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

   As previously stated PSE&G committed the Salem units to startup by dilution. Neither Westinghouse 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 1 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 Westinghouse have concluded that startup by dilution is not required in response to the Salem rod control system failure event.

Additional administrative controls for plant startup and power operation are not considered necessary since the General Operating Procedure for Reactor Startup contains procedural controls for the withdrawal of all shutdown and control rod banks. These controls require signoffs to ensure that all rod demand step counters are reset to zero prior to withdrawal and that all rod drive urgent failure alarms are reset. Additionally, procedural steps require verification of agreement between digital rod position and demand step counter positions. These verifications are performed at the 12 step and fully withdrawn positions for the shutdown banks and at each 50 step interval for control banks.

Annunciator response procedures give procedural guidance during power operations if a rod misalignment occurs. The parameters monitored by these annunciators provide comparisons between DRPI to average DRPI in a bank, rod to rod deviation (DRPI to DRPI), and rod to bank deviation (DRPI to Step Counter). The alarm setpoint for these comparisons is >12 steps. Detailed response guidelines for this condition exist within the offnormal procedure for Misalignment of Control Rods, and is referenced within the annunciator response procedures.

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 Westinghouse and the WOG have, at various times, recommended that licensees provide additional discussions, training, standing orders, etc. to ensure that their operators are aware of what transpired at Salem. The recommendations of the Westinghouse 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.

At Callaway Plant, Information Notice 93-46 and Generic Letter 93-04 are required reading for all Operations personnel and the Training Department will discuss the circumstances surrounding this event with the operators during requalification training. This training will be completed for all operating crews by September 20, 1993.

Summary of the Generic Safety Analysis Program

Introduction

As part of the Westinghouse Owners Group initiative, the WOG Analysis subcommittee is working on a generic approach to demonstrate that for all Westinghouse 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 subcritical and power conditions to demonstrate that 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 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 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 which 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 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 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, 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 Westinghouse plants, maximizes DNBR margins, is approved by the NRC, and is licensed for a number of Westinghouse 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 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 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 demonstrate 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 three different banks which is not possible) shows a non-limiting DNBR and is therefore bounded by at power cases. This result is applicable to all other 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 Overtemperature Delta-T reactor trip, no credit is assumed for the $f(\Delta I)$ penalty function. 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 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 $F\Delta H$ and the change in the maximum transient $F\Delta H$ 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 were generated for all Westinghouse plants.

Conclusion

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

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.