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SUBJECT: Responds to GL 93-04,"Rod Control Sys Failure & Withdrawal of Rod Cluster Assemblies."

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August 5, 1993

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
Project Directorate I-3
Washington, D.C. 20555

Subject: Response to Generic Letter 93-04
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Ref.(a): Letter from R. C. Mecredy (RG&E) to J. C. Linville (NRC), "Rod Control System Malfunction," dated July 2, 1993

(b): Letter from J. A. Widay (RG&E), to G. Lazarowitz (NRC), "Rod Control System Question Response," dated July 8, 1993

Dear Mr. Johnson:

Generic Letter 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies," issued on June 21, 1993 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 is requested to 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) and 2.) Subsequent correspondence between the Westinghouse Owners Group and the NRC resulted in schedular relief for Required Response 1.(a) and 1 (b) Part 1 (NRC Letter to Mr. Roger Newton dated July 26, 1993). Those portions of the required actions will now be included with the 90-day licensee response.

Rochester Gas and Electric (RG&E) hereby submits its response to the Generic Letter as it applies to Ginna station. This response summarizes the compensatory actions taken in response to the Salem rod control system failure event. Our response also provides a summary of the results of the generic safety analysis program conducted by the Westinghouse Owners Group and its applicability to Ginna Station. RG&E considers this action to be complete with respect to the 45 day required response to GL 93-04 (as amended by

A030

July 26 NRC letter to Mr. Roger Newton).

We note that pursuant to the events on June 29 and July 2, 1993 involving the rod control system at Ginna Station, additional surveillance, troubleshooting, and monitoring of the rod control system and components has been conducted. The details of the events and our subsequent actions have been submitted under References (a) and (b). Members of RG&Es and NRC staff have communicated a number of times by telecon in an effort to keep the NRC fully apprised. Since RG&E and Westinghouse were satisfied that the Ginna Station events were not the result of the same problem which occurred at Salem, this response letter does not reiterate our actions taken as a result of the Ginna station event.

I declare that the statements and matters set forth herein are true and correct to the best of my knowledge, information, and belief.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

GAH\294
Attachments

Subscribed and sworn to before me
on this 5th day of August, 1993

Marie C Villeneuve

MARIE C. VILLENEUVE
Notary, Public, State of New York
Monroe County
Commission Expires October 31, 1994

xc: Mr. Allen R. Johnson (Mail Stop 14D1)
Project Directorate I-3
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Ginna Senior Resident Inspector

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RESPONSE TO NRC GL 93-04

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 2 and any additional compensatory actions judged to be appropriate.

Request:
"Describe any compensatory short-term actions taken or that will be taken to address any actual or potential degraded or nonconforming conditions (see Generic Letter 91-18) such as:"

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 Westinghouse response provided in OG-93-42, dated July 2, 1993 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. The WOG and Westinghouse have concluded that increased frequencies in surveillance testing is not required or appropriate in response to the Salem rod control system failure event.

   A recommendation was made by Westinghouse for utilities to ensure that their surveillance testing will demonstrate rod control system operability and address maintenance troubleshooting. RG&E is including an additional caution regarding unexpected rod motion and subsequent troubleshooting in the rod control system surveillance procedure, rod drop surveillance procedure and startup procedure.

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 Ginna Station event, reported in our letter dated July 2, 1993, abnormal rod motion was terminated after only one step. 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, and Ginna Station has not modified its procedure during startup.

As a short-term compensatory response to the Ginna Station event, the control rods were placed in manual control.

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 discussion, 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.

   RG&E has placed the Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-007 on the Plan-of-the-Day (POD) for read and acknowledge for all operators. In addition, the NRC Information Notice 93-46 and the INPO Significant Event Notification No. 100 was placed on the POD to heighten operator awareness. The training organization will assess changes to the training program to include the Salem event to determine if those changes can effectively augment the current operator training on control rod malfunctions.

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 both 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 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 three different banks which is not possible) shows a non-limiting DNBR. 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 Over-temperature 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 can be generated for all Westinghouse plants.

Conclusion

Using this approach, the generic analyses and their plant-specific application demonstrate that for R.E. Ginna DNB does not occur for their 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 Code 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.