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SUBJECT: Forwards response to GL 93-04, "Rod Control Sys Failure & Withdrawal of Rod Control Cluster Assemblies," summarizing evaluations & compensatory actions taken in response to Salem rod control sys failure event.

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INDIANA
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AEP:NRC:1190
GL 93-04

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
RESPONSE TO GENERIC LETTER 93-04

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

August 5, 1993

Dear Dr. Murley:

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 Monday, June 21, 1993. Generic Letter 93-04 was addressed to all licensees with the Westinghouse Rod Control System (except Haddam Neck) 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 (GDC 25). 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 (Required Response 1.(b)), and within 90 days, provide a plan and schedule for long-term resolution (Required Response 2). 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 actions will now be included with the 90-day licensee response.

Indiana Michigan Power Company (I&M) hereby submits its response to the Generic Letter as it applies to Donald C. Cook Nuclear Plant Units 1 and 2. This response summarizes the evaluations and the compensatory actions taken by I&M in response to the Salem rod control system failure event. It also provides a summary of the generic safety analysis program conducted by the Westinghouse Owners Group and its applicability to Cook Nuclear Plant. The results of the analysis indicated that for Cook Nuclear Plant Units 1 and 2, departure from nucleate boiling (DNB) does not

occur for their worst-case asymmetric rod withdrawal. I&M 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 Mr. Roger Newton).

This letter is submitted pursuant to 10 CFR 50.54(f) and, as such, an oath statement is attached.

Sincerely,

E. E. Fitzpatrick

E. E. Fitzpatrick
Vice President

dr

Attachment

cc: A. A. Blind - Bridgman
J. R. Padgett
G. Charnoff
NFEM Section Chief
J. B. Martin - Region III
NRC Resident Inspector - Bridgman

STATE OF MICHIGAN)
COUNTY OF BERRIEN)

E. E. Fitzpatrick, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the forgoing Response to GENERIC LETTER 93-04 and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

E E Fitzpatrick

Subscribed and sworn to before me this 5th

day of August, 19 93.

Linda L. Boelcke
NOTARY PUBLIC

LINDA L. BOELCKE
Notary Public, Berrien County, MI
My Commission Expires April 1, 1996

ATTACHMENT TO AEP:NRC:1190

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)) 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 preventive maintenance procedures. Based on the response provided in Westinghouse Owners Group (WOG) letter OG-93-42, there was no perceived need to increase the frequency of testing on a permanent or generic basis. It is understood that 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.

   At Cook Nuclear Plant, during each refueling outage, the rod control system cabinets are cleaned and inspected. The cards are checked and, if necessary, replaced. There have been no identified failures with the cards associated with the preventive maintenance activities. Also, in review of the rod control system failure history as requested by the WOG, no logic cabinet card failures were found that could have led to an event like Salem's. Thus, it is not deemed necessary to modify our methods of preventive maintenance on the rod control system.

   The procedure used at Cook Nuclear Plant to perform the 31 day Technical Specification Surveillance requires the reactor operator to monitor the rod position indicators (RPIs) during the test. If a rod control system failure event such as Salem's were to occur, the reactor operator would be alerted by the RPIs incorrect tracking and would immediately halt rod movement. Also, there is considerable confidence in the rod control system; thus it is not considered necessary to increase our surveillance frequency or modify the procedure.

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

As stated in Generic Letter 93-04, the Salem units committed 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 one minute would be reasonable. In fact, as demonstrated during the R. E. Ginna event (July 1993), abnormal rod motion was terminated after only one step both in automatic and manual rod control. Thus, the WOG and Westinghouse have concluded that startup by dilution is not required in response to the Salem rod control system failure event.

At Cook Nuclear Plant, dilution to critical is used for the initial startup of a new cycle. In the event of a mid-cycle shutdown, the reactor is returned to critical using control rods. The procedures used at Cook Nuclear Plant to return a unit to critical provide requirements which would alert the reactor operator of a rod control system failure event such as Salem's. During shutdown bank withdrawal, rod movement is stopped at least every 100 steps to collect data for an inverse count rate ratio (ICRR or 1/m) plot. For withdrawal of the control banks, rod movement is stopped at least every 50 steps for the 1/m plot data collection. The entire rod withdrawal evolution is performed with rod control in manual, and a reactor operator verifying proper rod movement by monitoring the RPIs and demand counter. Also during the evolution, a reactor operator is continuously monitoring the source range monitor scaler/timer. This instrument would have a significant count rate increase if a rod control system failure were to occur that could cause the reactor to reach critical conditions unexpectedly. The reactor operators at Cook Nuclear Plant are rigorously trained to expect criticality at any time and how to respond if such an event were to occur. Thus, we are confident that if a rod control system failure were to occur, which could cause unexpected criticality, the reactor operators would quickly recognize and respond to the event such that exceeding any fuel design limit is highly unlikely.

Any time the reactor operator is required to move rods, the RPIs and demand counter are monitored to verify proper movement. If a rod control system failure such as Salem's were to occur, the reactor operator would quickly notice such an event. Also, when the reactor is above approximately 30% power, the rod deviation monitor is operable and is a useful tool to alert the reactor operator

of abnormal rod movement. If a rod's RPI signal were to deviate by more than ±12 steps from its demanded position, an annunciator would alert the reactor operator. The rod deviation monitor is part of our plant process computer (PPC) software which has proven to be highly reliable. However, if the PPC or rod deviation monitor is ever declared inoperable, its operability is re-established by the Computer Science Department. Furthermore, the PPC continuously monitors all of the inputs to the program for validity to ensure that it is operable. Thus, during power maneuvers or steady state operation, the reactor operator will be alerted to a rod control system failure prior to exceeding any fuel design limit.

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" -

At various times, both Westinghouse and WOG have 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 the Salem rod control system failure event.

Upon issuance of NSAL 93-007, the Cook Nuclear Plant Safety and Assessment Department reviewed the document and subsequently recommended that the operators review it. Shortly following the recommendation, the licensed operators were required to review the rod control system failure event at Salem through NSAL 93-007. Note that these actions occurred prior to receipt of Generic Letter 93-04. Also, the Cook Nuclear Plant Training Department is scheduled to include an item in the next Operational Review package, which is based on the events that occurred at Salem. The Operational Review package material is to be covered during Operations Requalification Training Period 6, which is scheduled to begin September 20, 1993. Furthermore, the Salem rod control system failure is scheduled to be covered with Instrumentation and Control personnel as part of their continuing training program. Thus, the Cook Nuclear Plant has proactively and aggressively addressed the Salem event with its personnel.

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 reference 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 Neutron Flux or Over-Temperature Delta-T (OTDT) protection signal. If the reactivity increase 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 OTDT protection system for the core. The radial power "tilt" may also affect the excore detector signals used for the High Neutron 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 WCAP-13803 (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 OTDT 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. Since the differences in plant designs are accounted for by the adjustment approach, plant-specific DNBR calculations can and have been generated for all Westinghouse plants.

## Conclusion

Using this approach, the generic analyses and the plant-specific application demonstrate that for Cook Nuclear Plant Units 1 and 2, 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 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.