ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9308100210 DOC.DATE: 93/08/05 NOTARIZED: YES DOCKET #
FACIL:50-261 H.B. Robinson Plant, Unit 2, Carolina Power & Light C 05000261
AUTH.NAME AUTHOR AFFILIATION
DIETZ,C.R. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

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

DISTRIBUTION CODE: A030D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
TITLE: Generic Ltr-93-04-Rod Control System Failure & Withdrawal of Rod Cont

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-1 PD	2	2			
INTERNAL:	NRR/DRCH/HICB	1	1	NRR/DSSA/SRXB	1	1
	~~REG FILE~~ 01	1	1			
EXTERNAL:	NRC PDR	1	1			

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 6 ENCL 6

R I D S / A D D S

CP&L

Carolina Power & Light Company

ROBINSON NUCLEAR PLANT
POST OFFICE BOX 790
HARTSVILLE, SOUTH CAROLINA 29551

AUG 05 1993

Serial: RNP/93-1837

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
TRANSMITTAL OF RESPONSE TO GENERIC LETTER 93-04

Gentlemen:

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 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 forty-five (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 ninety (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 actions will now be included with the ninety (90) day licensee response.

Carolina Power & Light Company (CP&L) hereby submits its response to the Generic Letter as it applies to the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR2). This response summarizes the compensatory actions taken by CP&L 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 HBR2. CP&L considers this action to be complete with respect to the 45 day required response to Generic Letter 93-04 (as amended by July 26, 1993 NRC letter to Mr. Roger Newton).

A030

Questions regarding this matter may be referred to Mr. D. B. Waters at (803) 383-1802.

Very truly yours,

Charles R. Dietz
Vice President
Robinson Nuclear Plant

RES:lst

Enclosure

I, C. R. Dietz, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of my information, knowledge and belief; and the sources of my information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Eleanor C. Chappell
Notary (Seal)

ELEANOR C. CHAPPELL NOTARY *** PUBLIC WAKE COUNTY, N.C.

My commission expires: 2/6/96

cc: Mr. S. D. Ebneter
Ms. B. L. Mozafari
Mr. W. T. Orders

RESPONSE TO NRC GL 93-04

Compensatory Actions

The purpose of this report 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 Westinghouse Owners Group (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.

   Operations Surveillance Test OST-011, Rod Cluster Control Exercise & Rod Position Indication, is performed bi-weekly to verify proper operation of both the full-length control rod drive mechanisms and the associated rod position indicating circuits. Both the frequency and the content of this procedure were reviewed and determined to be technically adequate to demonstrate proper functioning of both the Control Rod Drive circuitry and the Control Rod Position Indication circuitry.

   Engineering Surveillance Test EST-049, Rod Drive Mechanism Operation Testing, is performed at each refueling. The purpose of this surveillance test is to verify the proper timing of each Rod Control System slave cycler by taking current traces of the Lift Coil, Movable Gripper Coil and Stationary Gripper Coil during rod movement. This test is capable of detecting improper operation of the slave cyclers and no additional testing is required. If a malfunction in the slave cycler circuitry is detected during this testing, then appropriate maintenance will be performed at that time.

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 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 Westinghouse have concluded that startup by dilution is not required in response to the Salem Rod Control System failure event.

The procedures governing plant startup[1] and power operation[2] were reviewed and determined to be adequate. No additional administrative controls were required to alter the method of plant startup or power operation.

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 were 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.

The H. B. Robinson Training Unit has developed a Real-Time Training package on the Salem event which includes a review of the Westinghouse Nuclear Safety Advisory Letter as well as Generic Letter 93-04. This training, for all of the active licensed personnel, will be completed by August 16, 1993.

The Abnormal Operating Procedure for a Malfunction of a Reactor Control System was reviewed in light of the Salem event and was found to have adequate guidance for the detection of and response to a Rod Control System failure. No additional instructions were found to be necessary.

---

1 GP-003, Normal Plant Startup From Hot Shutdown To Critical
EST-050, Refueling Startup Procedure

2 GP-005, Power Operation

In addition to the above actions, plant management committed to the NRC to complete the actions recommended by both Westinghouse in Nuclear Safety Advisory Letter NSAL-93-007 and by the NRC in Generic Letter 93-04. The status of these actions is as follows:

a) Licensed operators should continue the normal process of verifying that rod motion is proper for required movement.

   Action: Verification of proper rod motion is a fundamental part of both licensed operator training and operating procedures. The need to verify proper rod motion in light of the Salem event has been reiterated to the licensed operators through Operations Directive 93-009.

b) Licensee should confirm the functionality of rod deviation alarms.

   Action: The computer generated rod deviation alarm has been verified to be functional and capable of detecting a rod deviation once it reaches the amount of misalignment specified in Technical Specifications.

c) Operators should review NSAL-93-007 to ensure their understanding of the event.

   Action: As discussed previously, training on the Salem event is being provided to all active licensed personnel.

This commitment is currently scheduled to be complete by August 16, 1993, when Item C is completed. In the event of a plant shutdown or trip prior to completion of these actions, the plant will not be restarted until all outstanding actions are completed.

Summary of the Generic Safety Analysis Program

Introduction

As part of the Westinghouse Owners Group (WOG) 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 (3-D) 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 OTDT 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 1 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 percent, 60 percent, and 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 HZP 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 (three 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 neutronic, transient, and subchannel thermal-hydraulic aspects of the Westinghouse Generic Safety Analysis Program are each applicable to the H. B. Robinson Steam Electric Plant, Unit No. 2.

With respect to core physics or 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 hot channel factor ($F_{\Delta H}$) and the change in the maximum transient $F_{\Delta H}$ from the symmetric and asymmetric RCCA withdrawal cases. The "3-loop, 5 rod D-bank" category is the specific A-factor applicable to the H. B. Robinson Steam Electric Plant Unit No. 2.

The three-dimensional (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 cases analyzed. The majority of these cases either did not generate a reactor trip or were terminated by a High Neutron Flux reactor trip. For the Overtemperature ΔT reactor trip, no credit is assumed for the f(ΔI) penalty function. The f(ΔI) penalty function reduces the OTΔT setpoint for highly skewed positive or negative axial power shapes. Compared to the plant-specific OTΔT setpoints that include the f(ΔI) penalty function, the setpoint used in the LOFT5 analyses is conservative, i.e., for those cases that tripped on OTΔT, a plant specific OTΔT 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 OTΔT are conservative for all Westinghouse plants.

With respect to subchannel thermal-hydraulic calculation of minimum DNB ratios, it is important to note that Siemens Power Corporation (SPC) supplies the fuel for the H. B. Robinson Steam Electric Plant, Unit No. 2. It is not practical for Westinghouse to provide precise estimates of the DNBR margin for fuel fabricated by SPC. Instead, Westinghouse will provide the core inlet water conditions or statepoints. In this way, the final step in the calculation process for the H. B. Robinson Steam Electric Plant, Unit No. 2 will be a SPC XCOBRA computer code calculation instead of a W THINC computer code calculation. While this process is not yet complete, the hydraulic design features of the SPC High Thermal Performance fuel (e.g., the presence of Intermediate Flow Mixing grids) provide a reasonable assurance that it is no more susceptible to DNB than fuel fabricated by Westinghouse.

Conclusion

Using this approach, the generic analyses and their plant-specific application demonstrate that for the H. B. Robinson Steam Electric Plant, Unit No. 2, DNB does not occur for their worst-case asymmetric rod withdrawal.

Additional-Short Term Action: Interim Plant Specific Safety Analysis by Siemens Power Corporation

Siemens Power Corporation (SPC) is in the process of reanalyzing Uncontrolled Withdrawal of a Single Full-Length RCCA at Full Power as the worst case currently presented in FSAR Section 15.4.3. Preliminary results show a calculated minimum DNBR above the safety limit based on:

- use of a nominal OTΔT trip setpoint of $K_1 = 1.1365$ instead of the more conservative usual analysis value of $K_1 = 1.24$.

- a less conservative, more specific estimation of the hot channel augmentation factor.

- use of the current applicable approved DNB correlation (i.e., ANFP instead of XNB).

Otherwise, the calculation basis for the case presented in the FSAR remains unchanged. It is a very conservative approach in that reactor power is assumed to increase to the OTΔT trip setpoint. No credit is taken for the High Neutron Flux reactor trip. The XCOBRA computer code is used to estimate the approach to DNB for a series of different conditions corresponding to the OTΔT setpoint.

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.