UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO THE CONTROL ROD DROP ANALYSIS
FOR
NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-282 AND 50-306

Introduction

By letter dated August 30, 1984, Northern States Power Company (the licensee) submitted Revision 2 of the reload methodology report titled "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units (NSPNAD-8102P Rev. 2 June 1984)." The staff completed the review of Revision 1 of the reload methodology report and the safety evaluation (SE) was issued by letter dated February 17, 1983. The SE contained several restrictions, one of which stems from the analysis of the control rod drop accident. Revision 2 of the methodology report NSPNAD-8102P (June 1984) gives the results of a reanalysis of the control rod drop accident which is the subject of this safety evaluation. Therefore, this safety evaluation updates section 2(d) of our safety evaluation issued on February 17, 1983 factoring in the results of the licensee's reanalysis. Other minor changes made in Revision 2 of the methodology report were also reviewed by the staff.

Evaluation and Conclusion

The safety evaluation in section 2(d) of our safety evaluation issued on February 17, 1983 is herewith changed to read as follows:

2(d) Control Rod Drop

A single or multiple dropped Rod Cluster Control Assemblies (RCCA) resulting from a single failure during power operation could result in a reduction in core thermal limits if the RCCA worth is too low to cause a negative flux rate trip. This problem occurs in the automatic mode of control operation when there is sufficient RCCA bank reactivity worth to allow the automatic control to withdraw the bank and raise the power to an overshoot condition while attempting to restore power to its original value. Therefore, the licensee has divided the analysis for an RCCA drop event into two parts.

First, a determination is made in which dropped rods will trip the negative flux rate scram system and thus require no further analysis. The DYNODE-P computer code is used for this analysis to determine the relationship between dropped rod reactivity worth and flux, specified in terms of relative tilt as seen by the excore detectors, consistent with a two out of four (with the worst single failure) excore detector rate trip system logic. A nominal negative flux rate trip setpoint of 5% reactor total power (RTP) with a time constant of 2 seconds is used. Including instrument uncertainties, the nominal setpoint corresponds to an analysis value of 6.9% of RTP with a time constant of 2 seconds. We find the assumptions and methods used for this first part of the rod drop event, i.e., the determination of which dropped RCCAs would cause a trip, acceptable. The important cycle specific physics parameters for this flux rate trip analysis are: (1) moderator temperature coefficient, (2) Doppler coefficient, (3) effective delayed neutron fraction, (4) prompt neutron lifetime, (5) dropped rod worth, and (6) excore tilt.

For those rods which do not cause a trip, the DYNODE-P code is used to determine the transient consequences. The DNB response is evaluated with the licensee's approved thermal margin methodology based on the COBRA IIIC/MIT code. The acceptance criteria for the event are that the DNBR calculated using the W-3 correlation is not less than 1.3 and that fuel temperature and cladding strain limits consistent with the acceptance criteria of SRP 4.2 are not exceeded. The limiting initial condition peaking factors for which limiting rod drop transients will not exceed DNB limits are determined. The relative tilt as seen by the excore detector is evaluated by correlating the core edge power densities to the excore detector readings. The excore detectors averaged response with the worst failure is input to the rod controller. Inserted control bank reactivity worth is calculated in three dimensions based on Technical Specification bank insertion limits. The least negative moderator and Doppler coefficients are used to maximize the transient power overshoot. In addition, the calculation is performed assuming full power with the most adverse combination of steady state errors applied to neutron flux level, coolant pressure, and inlet coolant temperature. We find these assumptions and analytical methods acceptable. The important cycle specific physics parameters for the rod drop transient with no trip are: (1) moderator temperature coefficient, (2) Doppler coefficient, (3) nuclear heat flux hot channel factor, (4) nuclear enthalpy rise hot channel factor, (5) effective delayed neutron fraction, (6) prompt neutron lifetime, (7) dropped rod worth, (8) control bank worth, and (9) excore tilt.

We have reviewed the accident definition, the analytical methods and assumptions used, the important physics parameters selected, the specific physics calculations performed each cycle, and the bounding values used in the safety analysis, and find them to be acceptable.

Based on the above, we conclude that the current operating restriction on the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 which requires the control rods to be manually operated at power levels above 90% when D bank control rods are less than 215 steps withdrawn can be removed from the plant operating procedures. In addition, this evaluation can serve as a basis for proposing appropriate changes to the technical specifications.

Additional minor changes that have been made in Revision 2 of the methodology report have also been reviewed. We have concluded that these minor changes in no way alter the staff's conclusions that were made in the SE issued February 17, 1983 and therefore are acceptable.

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