SAFETY EVALUATION REPORT BY THE OFFICE OF NRR

CONCERNING

INDUCED NEUTRON FLUX ERROR

FOR

BABCOCK AND WILCOX REACTORS

Introduction

In October 1980 Babcock & Wilcox (B&W) indicated (Ref. 1) that studies recently performed had concluded that event induced errors in the neutron flux detector readings and thus effective flux trip levels could be larger for some events than those normally assumed in analyses. The staff responded, following conversations with B&W, by requiring information from utilities (Ref. 2). The utilities with operating B&W reactors have responded (Ref. 3) and the response has been reviewed. The response and review are summarized here.

In brief the problems are (1) for some cooldown events the colder water in the downcomer region increases neutron flux attenuation thus potentially increasing the transient flux error on the excore nuclear instrumentation (ENI) beyond the 2% normally used in analysis, and (2) for control rod ejection events the neutron flux distribution change resulting from the abnormal control rod pattern causes effective levels in the excore detectors to change (for a given core average level). Both effects affect trip levels and potentially in an amount beyond that normally assumed.

All of the respc ding utilities, except Duke (Oconee reactors) presented a similar response, based on B&W calculations which were in turn primarily based on the calculations for the WPPSS-WNP 1/4 reactors which had initiated the problem concern. Duke carried out their own calculations and presented therefore, a somewhat different viewpoint. All concluded that the result of potential flux error increases were suitably bounded within the existing operating parameters of their respective reactors. In the following discussion the two presentations will be referred to as the B&W and Duke analyses. Both analyses were used in forming a judgment for the review.

Evaluation

Based on the WPPSS study B&W concluded that a limiting maximum overcooling event, among the small steamline break, feedwater and turbine bypass events, was a turbine bypass with peak inlet (and downcomer) temperature reduced by 16°F. They concluded that larger steamline breaks would be terminated by a building pressure or variable low pressure trip. Duke studied (analyzed) several overcooling events, including turbine bypass failure with ICS failure, and also studied the larger steamline breaks assuming a high flux trip was required.

Normally B&W has used a 2% transient flux error. This, along with other assumed errors and a trip setpoint of 105.5% of full power gives a trip in analyses of 112%. Based on ANISN calculations (from the WPPSS study) to translate downcomer temperature changes to ΔNI the maximum transient inlet temperature reduction of about 16°F corresponds to 13% ΔNI, giving an effective trip point of 123%. Duke examined data from a number of test programs relating temperature and flux readings. Based on these tests they developed a relationship (linear with temperature) between inlet temperature and ΔNI (at a 95% confidence level). It would provide a 12% ΔNI at 16°F. For much of their analysis, however, they used a 1% ΔNI/1°F factor (16% ΔNI at 16°F).

Using the calculated ΔNI vs inlet temperature relationship, B&W developed, for each reactor, at its minimum pressure (trip setpoint) a (graphical) relationship between reactor power, outlet temperature, trip lines (high flux with error and variable low pressure - outlet temperature) and thus regions protected by the reactor protection system. (This is best described in the Davis-Besse submittal). They superimposed on this DNBR values calculated using design power distributions. The results, which of course take advantage of the improved DNBR value at the lower inlet temperature conditions, demonstrate that DNBR limits (both 1.30 and 1.43 which includes a 10.2% rod bowing penalty) fall within the protected region for overcooling conditions out to, and beyond, 16°F overcooling. Power distribution calculation for 125% full power conditions were also done to check perturbations in distributions at these limiting conditions. These were also used to demonstrate margin to DNB and center fuel melt (CFM) limits.

Duke performed plant specific analyses for each overcooling transient, including the turbine bypass event (also giving the maximum overcooling as above) and the larger steamline breaks accidents (assuming a high flux trip is required). They used 1% ΔNI/°F to identify maximum (non trip) power levels (giving about 11% ΔNI for the turbine bypass) and assumed ICS failures to maximize overcooling and analyzed for DNB using design peaking factors. They found that DNB and CFM limits were not exceeded, even without the reduction which would have been provided by a lower trip level which would occur using the derived ΔNI - temperature error rather than 1% ΔNI/°F.

The review of all of the submittals has lead to the conclusion that the magnitude and extent of the effect and its consequences during events of interest have been suitably examined. The B&W calculations and the Duke measurements complement each other on the magnitude of ΔNI vs temperature as do the complementary calculations for the magnitude of temperature decrease to be considered during maximum events. Using this information the protection system appears to be able to provide protection before exceeding limits on DNB and CFM. However, all future submittals which require analysis of overcooling events by B&W reactors should include in the analysis and presentation an equivalent of the information involved in the present submittals and the use of the penalties resulting from inlet cooling similar to these unless new values are justified.

The other event involving a potential indication error for the flux signal, which in turn is involved in terminating the event by a trip signal, is the rod ejection accident. In this case the error arises from the change in power distribution caused by the ejected rod making the effective power level as seen by the flux detector different from the average used in (point kinetics) analyses. The problem, as related to trip, would only exist for small worth rods (neighborhood of 0.2% Δk or less) since the rise in flux level is too large to significantly affect trip occurrence and timing for larger reactivity insertions. Since the B&W "zero power" event analyses normally involve high pressure trips rather than high flux trip for smaller rod worths, the problem is only relevant to the full power analyses which are normally analyzed as tripping on high flux.

The B&W submittals argued on the basis of engineering judgment that, if heat transfer out of the fuel pin during the transient were included in the ejection analysis (as has not been the case in past submittals), the power and peaking increases for the range of reactivity insertion that might not cause flux trips would not result in peak enthalpies exceeding limits (280 cal/gm). Duke presented results of calculations of flux errors resulting from a number of rod configurations, providing a basis for a correlation of error with rod worth, and also presented typical power histories as a function of rod worth. From these it can be concluded that there would be a high flux trip for a rod worth above about 0.1% Δk at a trip level of about 120% (rather than the usually assumed 112%). For rods under this level there might not be a flux trip, however, power levels and peaking factors associated with these rod worths are sufficiently low that the limit for the event (280 cal/gm) is not approached. The initial transient is minor and the quasi-steady state is similar to that of the single rod withdrawal event. The latter is described in the Midland SAR where it is indicated, in an analysis with heat transfer, that 280 cal/gm is not approached (nor is DNB reached) for even larger rod worths than are involved here (e.g., greater than 0.3% Δk).

The review of the submittals has lead to the conclusion that the flux error associated with the changed power distribution for rod ejection does not significantly affect the trip function for the larger rod worth events and that the consequences for the smaller worth events are not of a magnitude to approach limits when considering the heat transfer that occurs.

Summary and Conclusions

The effective neutron flux trip level in B&W reactors may be raised above that normally used in analyses because of increased flux attenuation in the downcomer in cooldown events and because of power (flux) distribution changes in the rod ejection event. However, analyses of extreme cooldown events requiring high flux trip indicate that sufficient margin exists in the trip levels, as augmented by the improvement in DNBR provided by the cooldown, that limits on DNB and CFM are not exceeded in operating reactors. The review of this analysis has resulted in agreement with this conclusion for operating reactors. However, all future analyses of these events for B&W reactors should include in the effective trip level for cooldown events a suitable flux error term of a magnitude as discussed in this review, e.g. 13% ΔNI for a 16°F cooldown, or as specifically derived for the reactor as has been done by Duke. For the rod ejection event the analysis of the increased error indicates that the only events which may be significantly affected are those with smaller rod worths for which the consequences are below limits even without a high flux trip. The review has concluded that no changes are needed in operating parameters for currently operating reactors because of this error.

References

1. Letter from James Taylor (B&W) to Victor Stello (NRC), October 29, 1980: "Results of Recent Induced Flux Error Investigations."
2. Memorandum from L. S. Rubenstein (NRC) to T. Novak (NRC), November 28, 1980: "B&W Induced Flux Error."
3. Letters from the following utilities on the indicated dates to the NRC, Operating Reactor Branch 4.

Toledo Edison, March 18, 1981

Duke Power Co., March 19, 1981

Sacramento Municipal Utility District, March 20, 1981.

Metropolitan Edison Co., September 29, 1981.

Florida Power Corp., March 20, 1981

Arkansas Power & Light Co., January 30, 1981.