Attachment 1

Proposed Changed Pages

Unit 1	Revision
Page 3/4 2-8	Replace
Page 3/4 2-10	Replace
Page B3/4 2-4	Replace

Unit 2	Revision
Page 3/4 2-8	Replace
Page 3/4 2-10	Replace
Page B3/4 2-4	Replace

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55 \, [1 + 0.3 \, (1-P)]$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

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$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations:

a. Abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. Although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. Errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, completely offset any rod bow penalties. This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28
2) Axial Grid Spacing Coefficient (K_S) of 0.046 vs. 0.059
3) Thermal Diffusion Coefficient of 0.038 vs. 0.059
4) DNBR Multiplier of 0.865 vs. 0.88
5) Pitch reduction

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\,[1 + 0.3\,(1-P)]$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

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$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations:

a. Abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. Although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. Errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, completely offset any rod bow penalties. This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28
2) Axial Grid Spacing Coefficient (K_s) of 0.046 vs. 0.059
3) Thermal Diffusion Coefficient of 0.038 vs. 0.059
4) DNBR Multiplier of 0.865 vs. 0.88
5) Pitch reduction

ATTACHMENT 2

Significant Hazards Evaluation Pursuant to 10 CFR 50.92 for the Proposed Removal of the Rod Bow Penalty from the Technical Specifications

Proposed Change

Remove the Rod Bow Penalty (RBP) from the Units 1 and 2 Technical Specification 3.2.3 and revise the corresponding discussion in the bases.

Background

The phenomenon of fuel rod bowing in Westinghouse PWRs is considered in the departure from nucleate boiling ratio (DNBR) safety analysis of Farley Nuclear Plant Units 1 and 2. In the early 1970s much larger fuel rod bowing than had been predicted was observed in Westinghouse low parasitic (LOPAR) fuel assemblies. Due to this larger rod bow, the DNBR analyses were reevaluated. Subsequently, the DNBR effects due to rod bow were redefined and a rod bow penalty (RBP) was applied to the technical specification calculation of $F_{\Delta H}^{N}$. Additionally, the RBP as a function of burnup was defined in the Technical Specifications by the addition of Figure 3.2-3.

In an effort to gain insight into the extent and effect of fuel rod bow on DNBR Westi ;house Electric Corporation has obtained data from irradiated fuel assemblie: In 1975 sufficient rod bowing information was available to develop an empirical model to predict rod bow as a function of region average burnup. This information and the effects of predicted rod bowing on power peaking and DNBR analyses were presented in the original WCAP-8691, which was submitted for NRC review in January 1976. Revision 1 of WCAP-8691 (Reference 1) and References 2 and 3 document subsequent NRC inquiries and Westinghouse responses. The Westinghouse methods for predicting the effects of rod bow on DNB as described in the above documents were approved by the NRC staff in Reference 4.

WCAP-8691 Revision 1 and References 2 and 3 have successfully demonstrated that applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR and/or margin obtained from measured plant operating parameters, which are less limiting than those required by the plant safety analysis, can be used to offset the effect of rod bow. The maximum DNBR penalty which must be accounted for is $< 3\%$ at 33,000 MWD/MTU as identified in Reference 3. The safety analyses for Farley Nuclear Plant Units 1 and 2 maintained sufficient margin to accommodate full and low flow DNBR penalties. This margin totals 9.1% DNBR and includes the following:

	DNB Margin %
a. Design limit DNBR of [1.30 vs. 1.28],	1.6
b. Axial Grid Spacing Coefficient (K_s) of [0.046 vs 0.059],	2.9

		DNB Margin %
c.	Thermal Diffusion Coefficient of [0.038 vs 0.059],	1.2
d.	DNBR Multiplier of [0.865 vs 0.88], and	1.7
e.	Pitch reduction.	1.7
	Total:	9.1%

This margin is adequate to offset all rod bow penalty for assembly average burnups of up to 33,000 MWD/MTU. The maximum rod bow penalties accounted for in the design safety analysis are based on an assembly average burnup of 33,000 MWD/MTU. At burnups greater than 33,000 MWD/MTU, credit is taken for the effect of $F_{\Delta H}^{N}$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required.

References

(1) Skaritka, J., (Ed), "Fuel Rod Bow Evaluation," WCAP-3691, Revision 1 July 1979, (Proprietary).

(2) "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" letter, E. P. Rahe, Jr., (Westinghouse) to J. R. Miller (NRC), NS-EPR-2515, dated October 9, 1981, (Proprietary).

(3) "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" letter, E. P. Rahe, Jr., (Westinghouse) to J. R. Miller (NRC), NS-EPR-2572, dated March 16, 1982, (Proprietary).

(4) NRC letter from C. Thomas, NRC, to E. P. Rahe, Westinghouse dated December 29, 1982.

Analysis

Alabama Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed deletion of the rod bow penalty from the technical specification calculation of $F_{\Delta H}^{N}$ and considers the proposed change not to involve a significant hazards consideration. In support of this conclusion the following analysis is provided:

1) The proposed change will not significantly increase the probability or consequences of an accident previously evaluated because an adequate margin of safety to the minimum DNBR of 1.30 will be maintained for those transients which must account for the phenomenon of fuel rod bowing in the DNBR safety analysis (i.e., Condition I and Condition II events).

2) The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the plant configuration or mode of operation is not altered by the removal of the rod bow penalty. Adequate margin exists in the plant safety analysis to allow operation without the requirement for any DNBR related rod bow penalty.

3) The proposed change will not involve a significant reduction in margin of safety because the safety analyses for Farley Nuclear Plant Units 1 and 2 maintain sufficient margin to accommodate the removal of all rod bow penalties related to DNBR and the margin within the existing DNBR limits remains unaffected.

Conclusion

Based upon the analysis provided herewith, Alabama Power Company has determined that the proposed technical specification change will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, Alabama Power Company has determined that the proposed change meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

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