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DUKE POWER

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

Subject: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414
Supplement to Technical Specification Amendment Request
Unit 1 Cycle 7 Reload

Per discussions with your staff, additional information is being provided regarding the following items:

1) Fully withdrawn rod position varying between 222 steps withdrawn and 230 steps withdrawn, and

2) The assumption that 50% of fuel fails due to a rod ejection accident.

If there are any questions regarding the items above, contact Mary Hazeltine at (803)831-3080.

Pursuant to 10 CFR 50.91(b)(1) the appropriate South Carolina State official is being provided a copy of this amendment request.

Very truly yours,

M. S. Tuckman by

M. S. Tuckman

Attachments

MHH/C1C7.SUP

140106

A001

Ltr. Encl.

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RCCA Positioning

For Catawba 1 Cycle 7, the fully withdrawn RCCA position will vary from 222 steps withdrawn (SWD) to 230 SWD. The top of the active fuel corresponds to approximately 225 to 226 SWD. The neutronic behavior of the core will be affected by maintaining the RCCA's partially in the active fuel region.

The impact of the RCCAs being inserted to 222 SWD was evaluated in a 10 CFR 50.59 evaluation "RCCA Fully Withdrawn Operation at 222 steps or Above" dated October 8, 1991. This evaluation relied on a Reload Safety Evaluation (RSE) performed by Westinghouse on Catawba 2 Cycle 5. The conclusions of this RSE were that the various core physics parameters were determined to remain bounded by previous analyses.

Duke Power Company has also evaluated the impact of partially inserted control rods. The following parameters were examined:

- Axial Offset,
- Reactivity (i.e. Boron Concentration),
- Peaking Factors (F_Q and $F_{\Delta H}$),
- Shutdown Margin,
- Trip Reactivity, and
- Rod Worths.

The results of this analysis showed a negligible impact from positioning the RCCA's slightly in the active fuel region. In addition, Catawba 1 Cycle 6 recently repositioned their fully withdrawn RCCA position from 222 to 226 SWD. Predictions by Duke Power over-predicted the impact of withdrawing the RCCA's out of the active fuel. This implies that the impact of having the RCCA's positioned in the active fuel was being over-predicted. Therefore, since having the RCCA's partially inserted was determined to have a small effect on the important physics parameters, and the impact of the slightly inserted RCCA's was over-predicted, it can be concluded that positioning the RCCA's to 222 SWD will have a negligible effect on the neutronic behavior of the core. Additionally, it should be noted that all calculations used for predictions are performed at an intermediate rod position between 222 and 230 SWD. All safety related calculations are performed at the most conservative fully withdrawn position.

Rod Ejection Pin Failure

The radiological consequences of a postulated rod ejection accident must limit the offsite dose to an acceptable level. One of the assumptions for calculating the offsite dose is the number of failed pins due to departure from nucleate boiling (DNB). The methodology documented in Duke Power Company's topical report DPC-NE-3001 assumes that 50% of the fuel pins fail. As part of the analysis for each reload core, the number of pins experiencing DNB due to the rod ejection accident is calculated. The actual number of failed pins is then verified to be less than the 50% limit assumed in the dose analysis.

In the cycle specific analysis, a 3-D nodal code (EPRI-NODE-P) is used to calculate a 3-D assembly average power distribution. A separate code (PDQ-XD) is used to calculate the peak pin for each assembly. PDQ-XD calculates the peak pin to assembly average factor for the peak pin in each assembly. The assembly average power from the nodal code is multiplied by the pin to assembly factor for each assembly to obtain the peak pin for that assembly. To determine if a pin has exceeded the DNB limit, a thermal hydraulic analysis is performed using the computer code VIPRE-01 to calculate a Maximum Allowable Radial Peak (MARP). If the peak pin for an assembly exceeds the MARP, it is assumed to be failed. Since the pin to assembly factor is only for the peak pin in an assembly, it is conservatively assumed that if the peak pin fails, then all 264 pins in that assembly fail. Hence the actual number of failed pins is over predicted using this method. If the 50% failed fuel limit is not met using this method, then a more detailed pin count can be performed using the actual pin power distributions for each assembly. The percentage of failed pins can be reduced by approximately 20% at beginning of cycle and slightly less than 10% at end of cycle using the more detailed pin count.

Other conservatisms also exist which lead to the over predicting of the percentage of fuel pin failures. Two of the more significant conservatisms are as follows:

1) The gas gap between the fuel pellet and the fuel cladding is assumed to close for all fuel in 0.5 seconds, and

2) A highly skewed top peaked power distribution is assumed at the beginning of the rod ejection accident, and this power distribution is held constant for the duration of the accident.

Since an acceptable dose analysis is satisfied using a conservative failed pin calculation, this criteria of the ejected rod accident is clearly satisfied.

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