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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Responds to NRC 810112 ltr requesting addl info re Cycle 6 reload.All fuel assmblies in Cycle 6 core have been insp. All fuel assemblies in Cycle 5 core were insp before fuel shuffle for Cycle 6.No defective springs were observed.

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L.J.

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January 22, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. W. Reid, Chief
Operating Reactors Branch No. 4

Re: Oconee Nuclear Station
Docket No. 287

Dear Sir:

In response to your letter dated January 12, 1981, requesting information concerning the Cycle 6 reload at the Oconee Nuclear Station Unit 3, please find the attached.

With regard to the inspection of fuel assembly hold down springs, all fuel assemblies in the Oconee 3 Cycle 6 core have been inspected. All fuel assemblies in the Cycle 5 core were inspected prior to the fuel shuffle for Cycle 6, and no defective springs were observed. Additionally, the hold down springs of fuel assemblies with control rods were again inspected following the fuel shuffle for Cycle 6, and no defective springs were observed. The control rods were removed prior to the inspection. Based on the results of this inspection, the fact that hold down springs of the suspect heat of material (HT67C4XS) are not currently in use at Oconee, and the results of previous inspections, it is considered that further inspections of this type are not necessarily required.

Very truly yours,

William O. Parker, Jr.

JLJ:pw
Attachment

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Responses to Request
For Additional Information

Oconee Nuclear Station
Unit 3

Question 1

References 1 and 2 (attached) discuss the Babcock and Wilcox TAFY and TACO codes as used for fuel and ECCS analyses. We believe that the revised LOCA kW/ft limits given in Ref. 1 apply to the Oconee Nuclear Station Unit 3, Cycle 6 reload. Therefore,

(a) verify that the NRC assumptions as given in Ref. 2 are correct and applicable to Oconee 3 for the proposed cycle of operation.

(b) verify that the revised LOCA kW/ft limits as given in Ref. 1 will be used for the proposed cycle of operation and that the Duke December 22, 1980 submittal provides Technical Specification limits to reflect this use.

Response

(a) Yes, the NRC assumptions given in Ref. 2 are correct and applicable to Oconee 3, Cycle 6.

(b) Refer to a letter dated December 22, 1980 from William O. Parker to Harold R. Denton.

Question 2

The Oconee Unit 3, Cycle 6 Reload Report (Ref. 3) describes fuel rod design Analyses for cladding collapse (4.2.1), cladding stress (4.2.2), and cladding strain (4.2.3). For each of these sections, identify the analyses presented as:

(a) Bounded by conditions previously analyzed in the Oconee Unit 3 FSAR.

(b) Analyzed specifically for Cycle 6 conditions using methods and limits previously reviewed and approved by the NRC.

(c) Analyzed specifically for Cycle 6 conditions using generic methods or limits not previously reviewed by the NRC.

Those analyses identified as category (a) or (b) need not be discussed in the reload report. Analyses identified as category (c) should be discussed in sufficient detail to allow review and approval by the staff prior to restart.

Response

The analyses presented are either category (a) or (b).

Questions 3 and 4 were not provided.

Question 5

On November 9, 1979, a letter (Ref. 5) was issued to all operating light water reactors on the subject of cladding, swelling and rupture models used in the ECCS analysis. Verify that your January 8, 1980 response is applicable to Cycle 6 operation at Oconee 3.

Response

Duke Power Company's letter of January 8, 1980 is applicable to all units and all cycles for which the currently applicable ECCS evaluation model, including the TAFY code, was utilized. Since the ECCS and fuel performance bases for Oconee 3, Cycle 6 are the same as those previously approved, the January 8, 1980 letter is applicable to Oconee 3, Cycle 6.

Question 6

The initial density (95.0% T. D.) of the Batch 8 fuel is higher than that of the three previous batches (94% T.D.) used in the Cycle 6 core. Were all changes due to the higher-density fuel, such as those shown in Table 4-2, calculated with the methods described in Ref. 6 and 7? Were any of the more recent Babcock and Wilcox densification methods, such as Rev. 8, considered in these non-LOCA portions of the Cycle 6 safety analysis?

Response

Yes, all changes were due to the higher density fuel.

No, none of the more recent Babcock and Wilcox densification methods were considered.

Question 7

A Babcock and Wilcox report on control rod guide tube wear (Ref. 9) is currently under review by the staff. Is this report applicable to Oconee Unit 3, Cycle 6? Do you concur with the findings of this report?

Response

Yes, this report is applicable to Oconee Unit 3, Cycle 6.

Yes, Duke Power Company does concur with the findings of this report.

Question 8

Section 6 of the reload describes a rod bow DNBR penalty calculated specifically for Cycle 6 operation.

(a) Please provide the details of this calculation. Specifically, what were the values of:

K, $\sigma(\Delta c/c)$, $F(k/\sigma)$, $F(k/\sigma)_{.95 \times .95}$, $\bar{U}$,

σ_u, σ_δ, $\bar{\delta}$, $N_{\delta-1}$, σ_b, N_{b-1} and $DNBR_b$

(b) What is the magnitude of credit taken for the flow-area reduction hot channel factor?

(c) Are the margins used to offset the rod bowing DNBR penalty employed solely for this purpose? If not, please provide justification for using three margins more than once.

(d) Amend the basis of the technical specifications to identify each generic or plant-specific margin that has been used to offset the reduction in DNBR due to rod bowing. Also, reference either the source or approval of each generic margin.

Response

(a) Please find the attached curve and intermédiate points from Babcock and Wilcox used to obtain these values. (Attachment 1)

(b) The magnitude of credit taken for the flow-area reduction hot channel factor is 1%.

(c) Yes, the margins used to offset the rod bowing DNBR penalty are employed solely for this purpose.

(d) The generic margin is included in the attached (Attachment 2) proposed revision to Oconee Technical Specification page 2.1-3d.

The Oconee 3, Cycle 6 specific margin was discussed in a previous Duke submittal dated June 14, 1978 which provided Amendment 1 (06/12/78) to Oconee 3, Cycle 4. The potential effect of fuel rod bow on DNBR was considered by incorporating suitable margins into DNB-limited core safety limits and RPS setpoints. The maximum rod bow penalty was calculated from the equation:

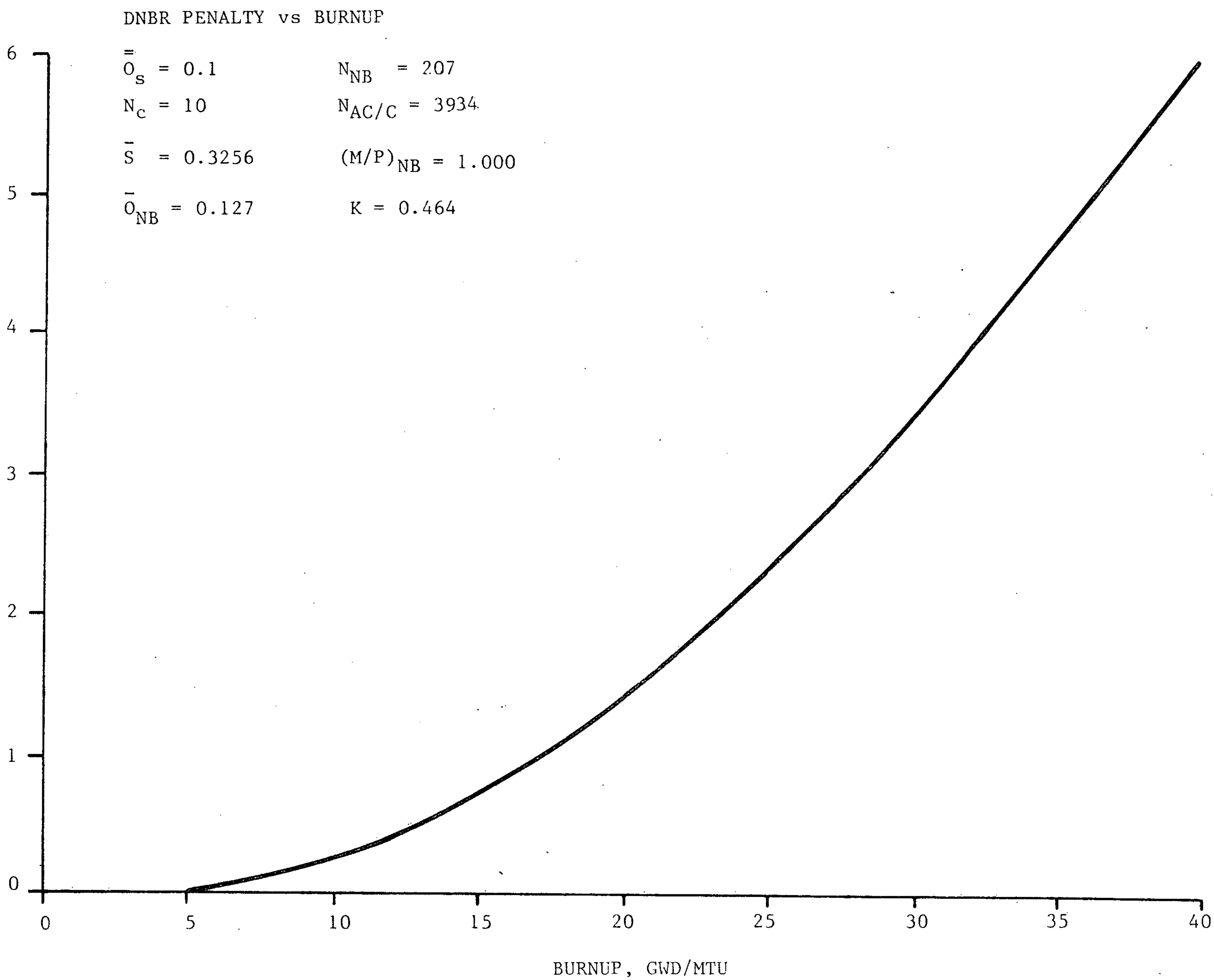
$$\frac{\Delta c}{Co} = 0.065 + 0.001449 \sqrt{BU}$$

where ΔC = rod bow magnitude, mils
Co = initial gap (138 mils)
BU = Maximum Assembly burnup, MWD/MTU

An 11.22% rod bow penalty based on an assumed burnup of 33000 MWD/MTU is applied to all analyses that define plant operating limits and to design transients. No fuel assembly will achieve a burnup as high as 33000 MWD/MTU during Cycle 6 operation. A thermal margin credit equivalent to 1% DNBR to offset the rod bow penalty has been used as a result of the flow area (pitch) reduction factor included in all thermal-hydraulic analysis. The 1% DNBR is the only credit applied to offset the rod bow penalty.

As stated in the Oconee 3, Cycle 6 reload report BAW-1634, page 6-1, the highest burnup expected is 23,411 MWD/MTU. The plant operating limits are based on a minimum 10% DNBR margin which is calculated by the above with the thermal margin credit applied.

ATTACHMENT 1


DNBR PENALTY vs BURNUP
$\bar{\bar{O}}_s$ = 0.1
N_{NB} = 207
N_c = 10
$N_{AC/C}$ = 3934
$\bar{S}$ = 0.3256
$(M/P)_{NB}$ = 1.000
$\bar{O}_{NB}$ = 0.127
K = 0.464
0
1
2
3
4
5
6
0
5
10
15
20
25
30
35
40
BURNUP, GWD/MTU

The rod bow DNBR penalty as a function of burnup is plotted on the attached curve. This curve was generated by the method described in Reference 1. The only change is that the value for K used in the sample calculation is 0.55 and the value for K used in the calculations generating the curve is 0.464. The value for K was calculated by the method outlined in Reference 2 using a DNBR penalty of 5.2% at 55 percent closure as described in Reference 3. The attached figure is a plot of the rod bow DNBR penalty vs. burnup and includes a summary of the parameters used in the calculation.

References: 1) Letter, J. H. Taylor, B&W to D. B. Vassallo, USNRC, dated December 13, 1978.

2) Letter, L. S. Rubestein, USNRC, to J. H. Taylor, B&W, dated October 18, 1979.

3) Letter, J. H. Taylor, B&W to S. A. Varga, USNRC, dated June 22, 1979.

ATTACHMENT 2

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 3.

Power peaking is not a directly observable quantity, and, therefore, limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup independent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis.(4)

The specified flow rates for Curves 1, 2 and 3 of Figure 2.1-2C corresponds to the expected minimum flow rates with four pumps, three pumps and one pump in each loop, respectively.

The maximum thermal power for three-pump operation is 87.2 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.08 = 80.7 percent power plus the maximum calibration and instrument error (Reference 4). The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For each curve of Figure 2.1-3C a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The curve of Figure 2.1-1C is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3C.

References

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March 1970.

(2) Oconee 3, Cycle 3 - Reload Report - BAW-1453, August, 1977.

(3) Amendment 1 - Oconee 3, Cycle 4 - Reload Report - BAW-1486, June 12, 1978.

(4) Oconee 3, Cycle 6 - Reload Report - BAW-1634, August, 1980.