7808/23/78

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FURNISHING CLARIFICATION TO APPLICANT"S 07/10/78 PROPOSED TECH SPEC CHANGES RE ALLOWING FULL PWR OPERATION WITH UP TO 25% OF THE STEAM GENERATOR TUBES PLUGGED, PERTAINING TO THE REACTOR CORE THERMAL HYDRAULIC LIMITS...W/ATT SUPPORTING FIGURES.

PLANT NAME: TURKEY PT #3
TURKEY PT #4

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REGULATORY DOCKET FILE COPY

FPL
FLORIDA POWER & LIGHT COMPANY

August 16, 1978
L-78-271

Office of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Jr., Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
<u>Thermal and Hydraulic Safety Limits</u>

Florida Power and Light Company (FPL) letter L-78-230 of July 10, 1978 contained proposed changes to the Technical Specifications of Turkey Point Units 3 and 4 to allow full power operation with up to 25% of the steam generator tubes plugged. The staff has asked FPL to clarify the derivation of the Reactor Core Thermal Hydraulic Limits, Figure 2.1-1b, which are to be added to the Technical Specifications to permit operation with steam generator tube plugging between 19% and 25%.

The basic methods used to produce this figure are described in FPL report NAD-QR-25, submitted with letter L-77-106 on April 4, 1977. A subchannel of the fuel assembly consisting of a RCCA guide thimble and 11 surrounding fuel rods was modeled by means of the COBRA IIIC thermal-hydraulics code; the geometry is shown in Figure 1. Of the eight subchannels, channels one and five contain an allowance for a water gap between adjacent assemblies.

A number of COBRA computer runs were made with the following input. For power levels $\geq$ 100% of rated the $F^N_{\Delta H}$ for each fuel rod was 1.55, while for lower power levels $F^N_{\Delta H}$ was taken to be the highest value permitted by the Technical Specifications,

$$F^N_{\Delta H} = 1.55[1 + 0.2(1-P)]$$

where P is the fractional power. The axial power distribution for all fuel rods was a center peaked chopped cosine with a peak to average ratio of 1.55. Fuel densification was accounted for by a reduction in active fuel length from 144 to 142 inches.

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Cross flow mixing parameters and flow correlations were taken to be those listed in Table 4 of NAD-QR-25. In accordance with the methods described in that report the core coolant average mass velocity of 2.32 * 10^6 lb/hr ft^2 was reduced by 5% to allow for inlet flow maldistribution. Flow was reduced by another 1% to account for flow redistribution and the effects of interassembly mixing. To generate Figure 2.1-1b the coolant flow was further reduced by another 5% to account for the effects of the 25% steam generator tube plugging. Appropriate uncertainty factors were included for power levels, pressures and temperatures.

The curves of Figure 2.1-1b, covering a range of pressure levels, were generated by varying the average coolant temperature for a given power level until the minimum DNBR, as calculated with the COBRA model described above, was 1.24 with the rod bow penalty included. Thus each curve represents the locus of points of the DNBR limit. The DNBR was calculated with the W-3 correlation and the "L" grid correction for LOPAR 15x15 Westinghouse fuel. The DNBRs obtained from the COBRA computer runs were reduced by 28.9% to account for the rod bow penalty prescribed for fuel with the highest burnup.

At the lower power levels the average coolant temperature at the reactor exit reaches the saturation temperature before the DNB limit is reached. The saturation line then represents the limiting condition. Figure 2 shows these saturation limit curves as well as the DNB limit curves. To be consistent with the Reactor Core Thermal and Hydraulic Safety Limit curves presently in the Technical Specifications and those submitted for less than 25% tube plugging on June 22, 1978 (letter L-78-217), horizontal limit lines are conservatively prescribed in Figure 2.1-1b instead of the higher saturation limit lines.

Very truly yours,

Robert E Uhrig

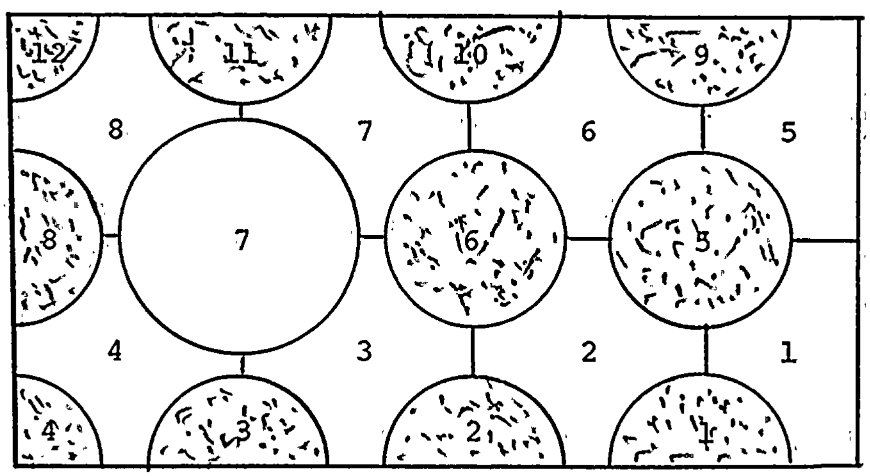
Robert E. Uhrig
Vice President

REU:RDH:lc

cc: Mr. Robert Lowenstein, Esquire
Mr. James P. O'Reilly, Region II

FIGURE 1

SUBCHANNEL GEOMETRY



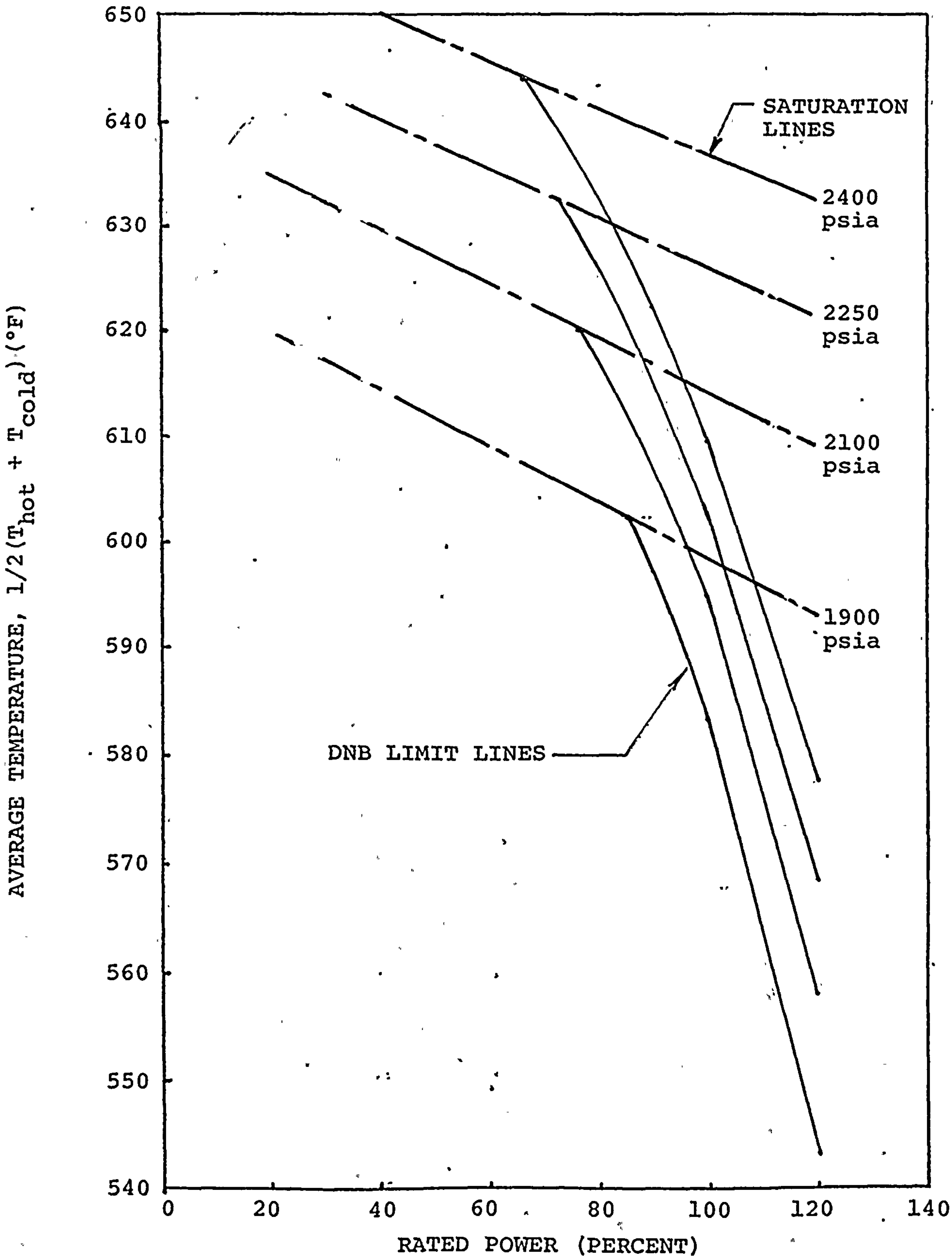


Figure 2. Reactor Core Thermal and Hydraulic Safety Limits