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AUTH.NAME AUTHOR AFFILIATION
URIG,R.E. FLORIDA POWER & LIGHT CO.
RECIP.NAME RECIPIENT AFFILIATION
STELLO,V. DIVISION OF OPERATING REACTORS

SUBJECT: SUBMITS ANALYSIS SUPPORTING AMEND TO APP A OF OL DPR-31 & DPR-41.DETERMINES DNBR SAFETY LIMIT OF 1.24 CONSISTENT W/COBRA IIIC COMPUTER CODE.FORWARDS "DNBR SAFETY LIMIT FOR COBRA III C ANALYSIS."

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FLORIDA POWER & LIGHT COMPANY

May 21, 1979
L-79-127

Director 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 & 4
Docket Nos. 50-250 and 50-251
Proposed Amendment to Facility
Operating Licenses DPR-31 and DPR-41

In accordance with 10 CFR 50.30, Florida Power & Light Company (FPL) submits herewith three (3) signed Originals and forty (40) copies of a request to amend Appendix A of Facility Operating License DPR-31 and DPR-41. The request has been prepared in response to a September 15, 1978 letter from Mr. A. Schwencer of your staff.

As requested in the September 15 letter (Reference 1), FPL has performed an analysis to derive a DNBR safety limit consistent with the COBRA IIIC computer code. This code was used in analysis submitted previously (References 2,3,and 4) to demonstrate the thermal margin available at Turkey Point Units 3 and 4 to compensate for a reduction in DNBR due to the effects of fuel rod bowing.

The NSSS vendor had obtained a DNBR limit of 1.24 from an analysis of critical heat flux tests using the THINC computer code with the W-3 critical heat flux correlation and the L-grid correction. However, this data base is proprietary and is not available to FPL. Instead, a corresponding DNBR limit was derived for the COBRA IIIC code by comparing the COBRA IIIC results with THINC results for a somewhat different data base (Reference 5) which was proposed in your letter of September 15. The analysis (attached) yielded a DNBR safety limit of 1.273 for the COBRA IIIC code. Revised "Reactor Core Thermal and Hydraulic Safety Limit" curves for tube plugging levels of $\leq$15%, >15% to $\leq$19%, and >19% to $\leq$25%, based on the derived DNBR limit of 1.273, are attached. Revised Bases pages are also attached.

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Mr. Victor Stello, Jr., Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission

The information supplied herewith supplements our previous submittals concerning the effects of rod bow on DNBR at Turkey Point Units 3 and 4. It is proposed that the attached Reactor Core Safety Limits be substituted for the present Figures 2.1-1, 2.1-1a and 2.1-1b in the Technical Specifications, and that with these new and more restrictive limits, the units can be operated safely without any other penalties for rod bow.

The proposed amendment has been reviewed by the Turkey Point Plant Nuclear Safety Committee and the Florida Power & Light Company Nuclear Review Board. They have determined that it does not adversely affect the health and safety of the public.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU:MAS:ncr

Attachment

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

REFERENCES

1. Letter, A. Schwencer to R. E. Uhrig, September 15, 1978, (Docket Nos. 50-250 and 50-251).

2. Margins in Turkey Point Units 3 and 4 Safety Analysis to Offset the Effects fo Fuel Rod Bowing, FPL Report NAD-QR-25, submitted with letter L-77-106, April 4, 1977.

3. Letter L-78-217, R. E. Uhrig to V. Stello, June 22, 1978.

4. Letter L-78-230, R. E. Uhrig to V. Stello, July 10, 1978.

5. Rosal, E. R. et al., High Pressure Rod Bundle DNB Data with Axially Non-Uniform Heat Flux, Nuclear Design Vol 31, 1974.

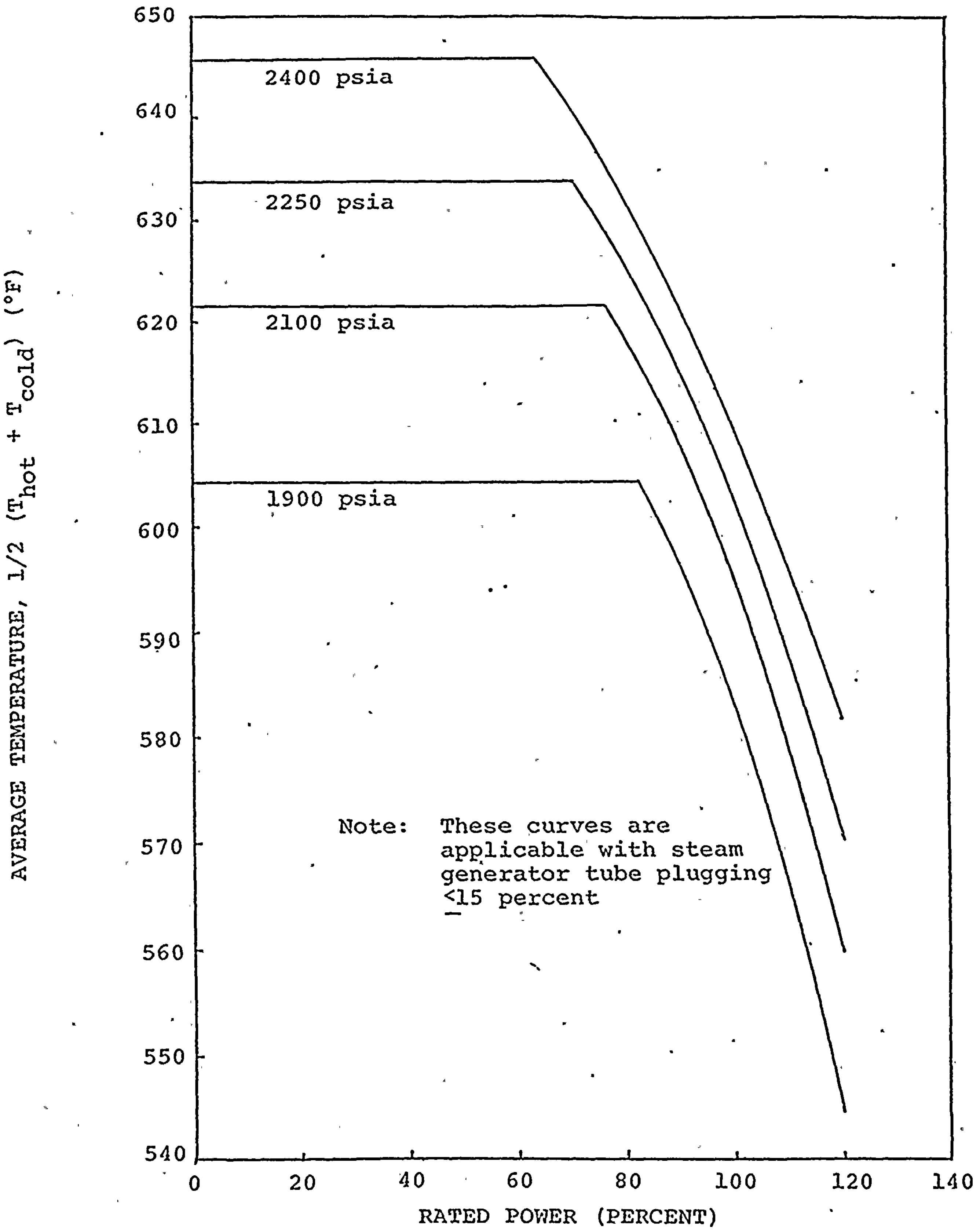

Figure 2.1-1 Reactor Core Thermal and Hydraulic Safety Limits, 3 Loop Operation

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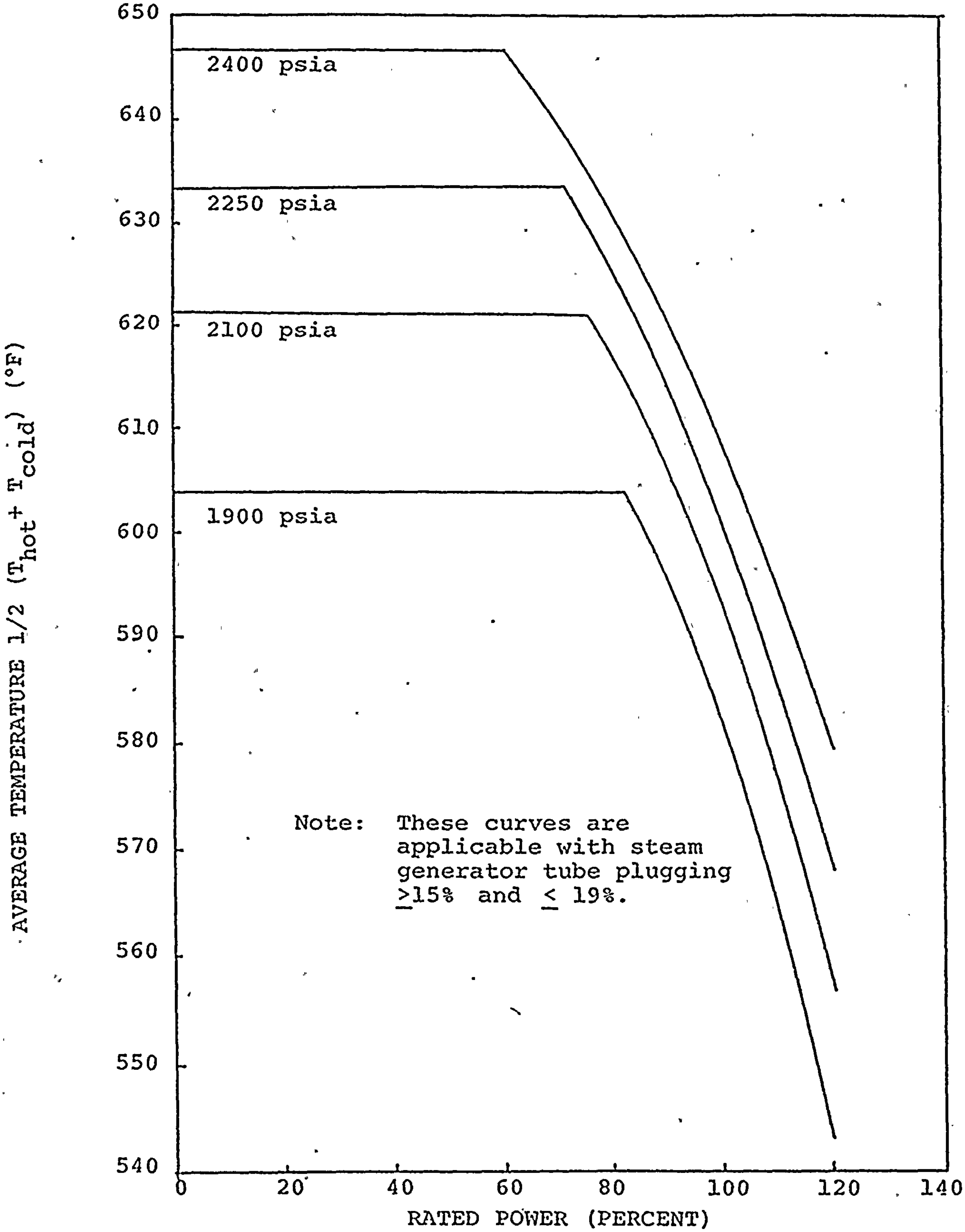

Figure 2.1-1a. Reactor Core Thermal and Hydraulic Safety Limits, 3 Loop Operation

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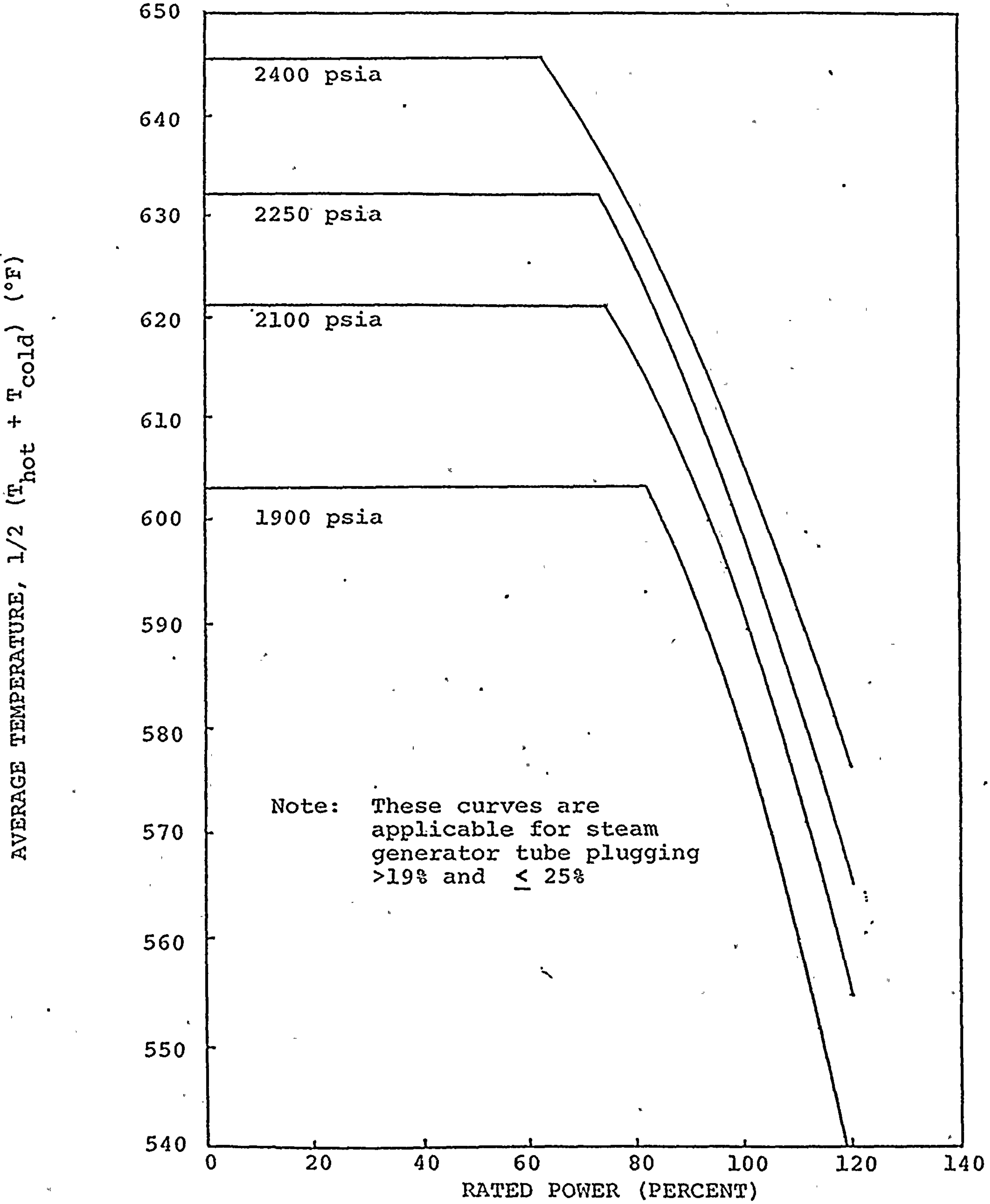

Figure 2.1-1b Reactor Core Thermal and Hydraulic Safety Limits, 3 Loop Operation

5/21/79

B2.1 BASES FOR SAFETY LIMIT, REACTOR CORE

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure; have been related to DNB through the W-3 DNB correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.273. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appriate margin to DNB for all operating conditions. (1,2)

The curves in the Specification represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNBR is no less than 1.273. The area of safe operation is below these lines.

The curves are based on the following nuclear hot channel factors:

$$F_q^N = 2.41$$

$$F_{\Delta H}^N = 1.55$$

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.2. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure 3.2-1 ensure that the DNBR is always greater at partial power than at full power.

The hot channel factors are also sufficently large to account for the degree of malpositioning of part-length *rods that is allowed before the reactor trip set points are reduced and rod withdrawal block and load runback may be required. (3) Rod withdrawal block and load runback occur before reactor trip setpoints are reached.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in exceeding DNBR design limits, including the effects of fuel rod bowing. (1) (2)

References

(1) NAD - 2199
(2) NAD - QR - 25
(3) FSAR 3.2.2.

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* Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.