CHANGE REQUEST NO. 15

PROVISIONAL OPERATING LICENSE DPR-16 (DOCKET 50-219)

Applicant hereby requests the Commission to change the Technical Specifications (Appendix A of the above captioned license) as follows:

1.A Specifications to be Changed

- Specification 2.1.A - Figure 2.1.1, Specification Bases 2.1.

1.B Extent of Changes

1. Specification 2.1.A, Figure 2.1.1 - Reevaluate the safety limit curves based on the XN-1 Minimum Critical Heat Flux Correlation and a peaking factor of 3.01.
2. Specification Bases 2.1 - Revise the bases to reflect the use of the XN-1 correlation and the new safety limits relating to fuel cladding integrity.

1.C Change Requested

1. Revise page 2.1-2 to read as shown in Attachment A.
2. Page 2.1-3, first paragraph, change the total peaking of 3.03 to 3.01 wherever it appears in the paragraph (lines 2, 11, 15 and last line).
3. Page 2.1-5, change reference (1) to read as follows:
   (1) L. H. Steves, A. M. Sutey, O. E. Fitzsimmons, "XN-1 Critical Heat Flux Correlation for Boiling Water Reactor Fuel".
4. Page 2.3-1, and 2.3-2, Limiting Safety System Settings, change total peaking factor of 3.03 to 3.01 (sections 1a and 2a).
5. Page 2.3-4, Paragraph 3, lines 2, 8 and 9, change 3.03 to 3.01.
6. Page 2.1-6. Revise Figure 2.1.1, Fuel Cladding Integrity Safety Limit.

ATTACHMENT A

A critical heat flux occurrence results in a decrease in heat transferre from the clad, and, therefore, high clad temperatures and the possibility of clad failure. However, the existence of a critical heat flux occurrence is not a directly observable parameter in an operating reactor. Furthermore, the critical heat flux correlation data which relates observable parameters to the critical heat flux magnitude is statistical in nature. Therefore, the safety limit represented in Figure 2.1.1 is based on the significant parameters involved in the critical heat flux correlation and is taken at the core design critical heat flux correlation level of confidence that a critical heat flux occurrence will not occur on any rod in the core.

The safety limit curves shown in Figure 2.1.1 represent the conditions f which there is 99 percent confidence that the most limiting rod has a minimum critical heat flux ratio (MCHFR) greater than 1.0. The MCHFR value was deter using the design basis critical heat flux correlation given in JN-72-18 (1). The operating range with MCHFR >1.0 is below and to the right of these curves

The design basis critical heat flux correlation is based on an interrelationship of reactor coolant flow and steam quality. Steam quality is determined by reactor power, pressure, and coolant inlet enthalpy, which, in turn, is a function of feedwater temperature and water level. This correlation is based upon experimental data taken over the entire pressure range of interest in a BWR, and the correlating line was determined by the statistical mean of the experimental data.

Curves are presented for two different pressures in Figure 2.1.1. The upper curve is based on a nominal operating pressure of 1035 psia. The lower curve is based on a pressure of 1250 psia. In no case is reactor pressure ever expected to exceed 1250 psia because of protection system settings well below this value, and, therefore, the curves will cover all operating conditio

with interpolation for pressures between 600 psia (the lower end of the critic
heat flux correlation data) and 1035 psia, the upper curve is applicable with
increased margin.

1.D Discussion

Exxon Nuclear Company is the fuel manufacturer for the Cycle III reload. This request for approval of a Technical Specification change to allow the use of the XN-1 correlation is based upon the formulation and use of this correlation by the Exxon Nuclear staff in their analyses. The XN-1 correlation is an extensive experimental evaluation of the critical heat flux (CHF) and hydraulic performance of commercial boiling water reactor fuel designs. It is the result of a parametric evaluation of 298 Exxon Nuclear 16 - rod bundle CHF data points and is shown to conservatively apply to an additional 310 rod-bundle CHF data points for BWR bundles presented in the literature. Only data representative of operating BWR rod bundles has been considered in this study. A complete description of the correlation is presented in reference (1) on page 2.1.5 of the Technical Specifications as revised in this submittal. Further reference to the correlation can be found in Facility Change Request No. 4 and its Supplement.

The maximum permitted local peaking factor for the CYCLE III reload fuel is slightly lower than during previous cycles. This offsets the increase in the average linear heat generation rate produced by changing the spacer capture rod to an inert rod in the 148 reload assemblies. The result is a total peaking factor reduction from 3.03 to 3.01. This is discussed in Section IV and shown in Table IV of Supplement No. 1 to Facility Change Request No. 4. Previous cycle data are presented in Table V of Facility Change Request No. 4.

JERSEY CENTRAL POWER & LIGHT COMPAN

By Ivan R. Finfrock Jr.
Vice President

STATE OF NEW JERSEY)
)
COUNTY OF MORRIS)

Sworn and subscribed before me this 10th day of May 19

John J. Prente
Notary Public

NOTARY PUBLIC OF NEW JERSEY
My Commission Expires December 9, 1976

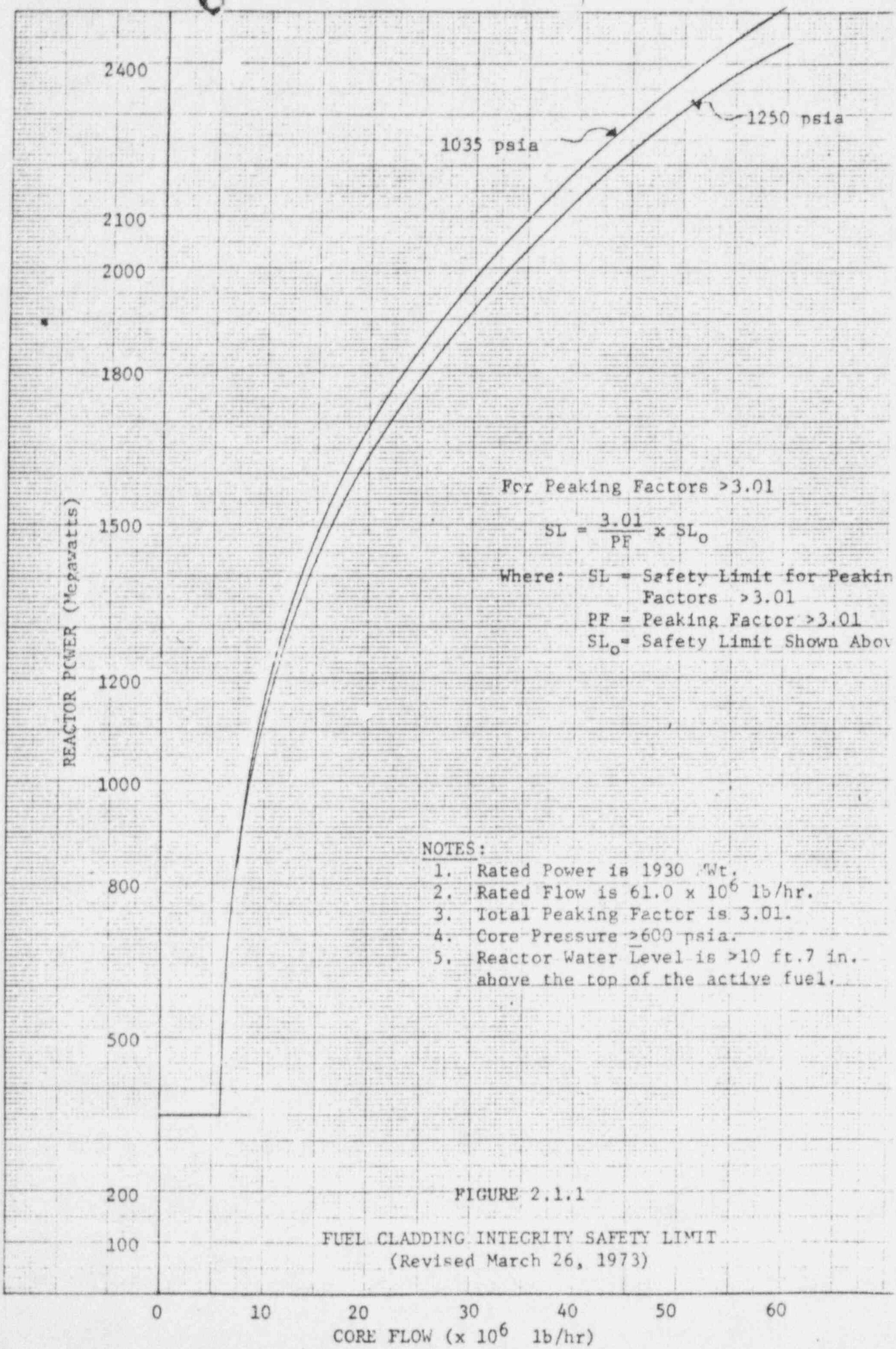

FIGURE 2.1.1

FUEL CLADDING INTEGRITY SAFETY LIMIT
(Revised March 26, 1973)