DUKE POWER COMPANY
OCONEE NUCLEAR STATION

ATTACHMENT

PROPOSED TECHNICAL SPECIFICATIONS

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LIST OF FIGURES

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The maximum local fuel pin centerline temperature shall be less than 5080 - $(6.5\times10^{-3})\times($ Burnup, MWD/MTU$)$ $^{\circ}$F. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits as specified in Figure 2.1-2.

The DNBR shall be maintained greater than the correlation limits of 1.3 for BAW-2 and 1.18 for BWC. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits and variable low RCS pressure limits as specified in Figures 2.1-2 and 2.1-1 respectively.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions and anticipated transients. This is accomplished by operating within the nuclear boiling heat transfer regime where the heat transfer coefficient is large and the cladding temperature is only slightly greater than the coolant temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation, but neutron power and reactor coolant pressure and temperature can be related to DNB using a critical heat flux (CHF) correlation. 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 actual local heat flux, is indicative of the margin to DNB.

The BAW-2 and BWC CHF correlations (1,2) have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The curve presented in Figure 2.1-1 represents the conditions at which the minimum allowable DNBR is predicted to occur for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based upon the design nuclear peaking factors provided in the Core Operating Limits Report.

Since power peaking is not a directly measurable quantity, DNBR limited power peaks and fuel melt limited power peaks are separately correlated to measurable reactor power and power imbalance. The reactor power imbalance limits, Figure 2.1-2, define the values of reactor power as a function of axial imbalance that correspond to the more restrictive of two thermal limits - MDNBR equal to the DNBR limit or the linear heat rate equal to the centerline fuel melt limit.

The core protection safety limits are based on an RCS flow less than or equal to 385,440 gpm (4 pump operation). Three pump operation is analyzed assuming 74.7 percent of four pump flow. The maximum thermal power for three pump operation is provided in Figure 2.1-2.

References

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

(2) Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143P, Part 2, August 1981.

Figure 2.1-1

Variable Low Pressure

Protective Limits

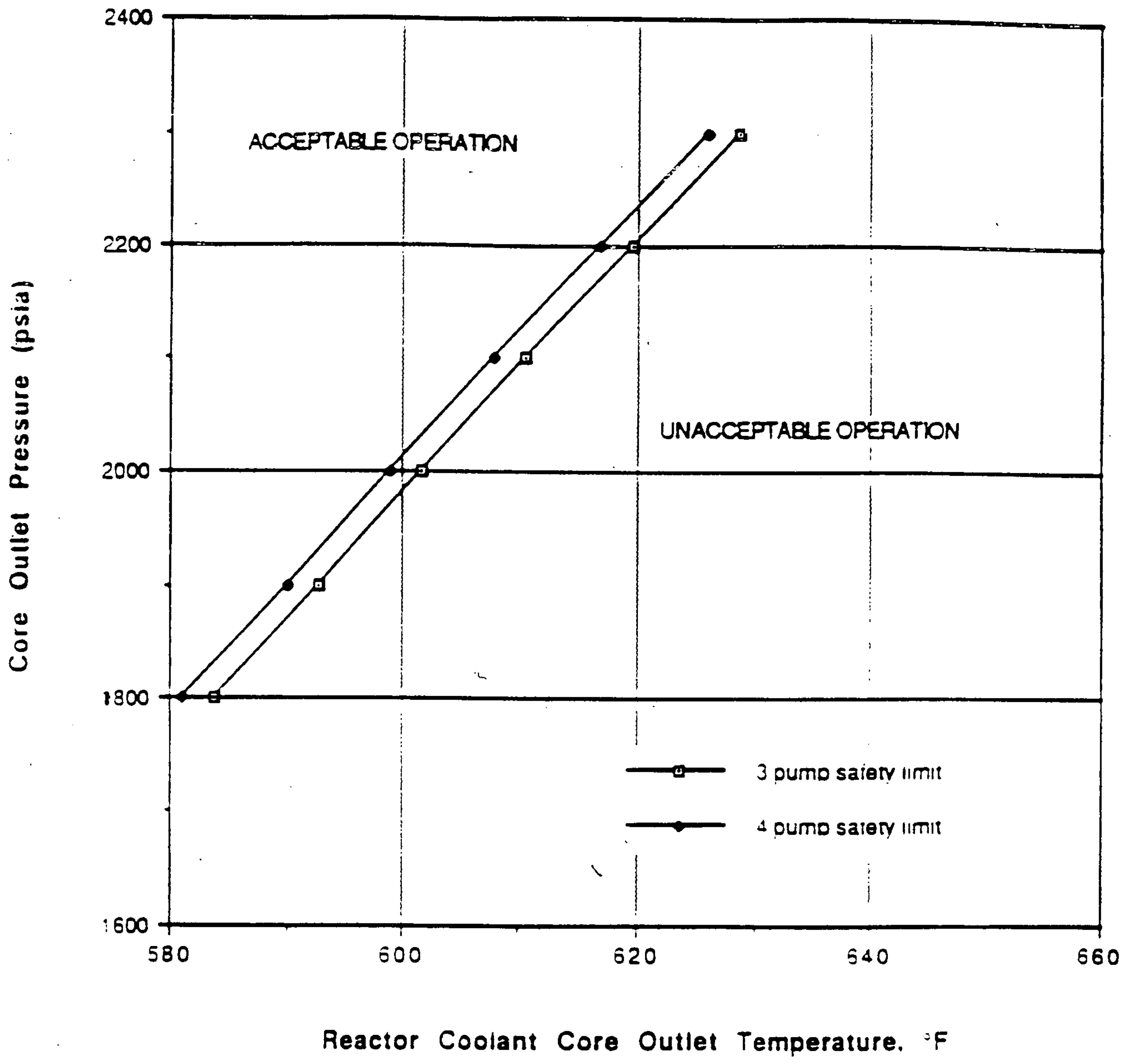
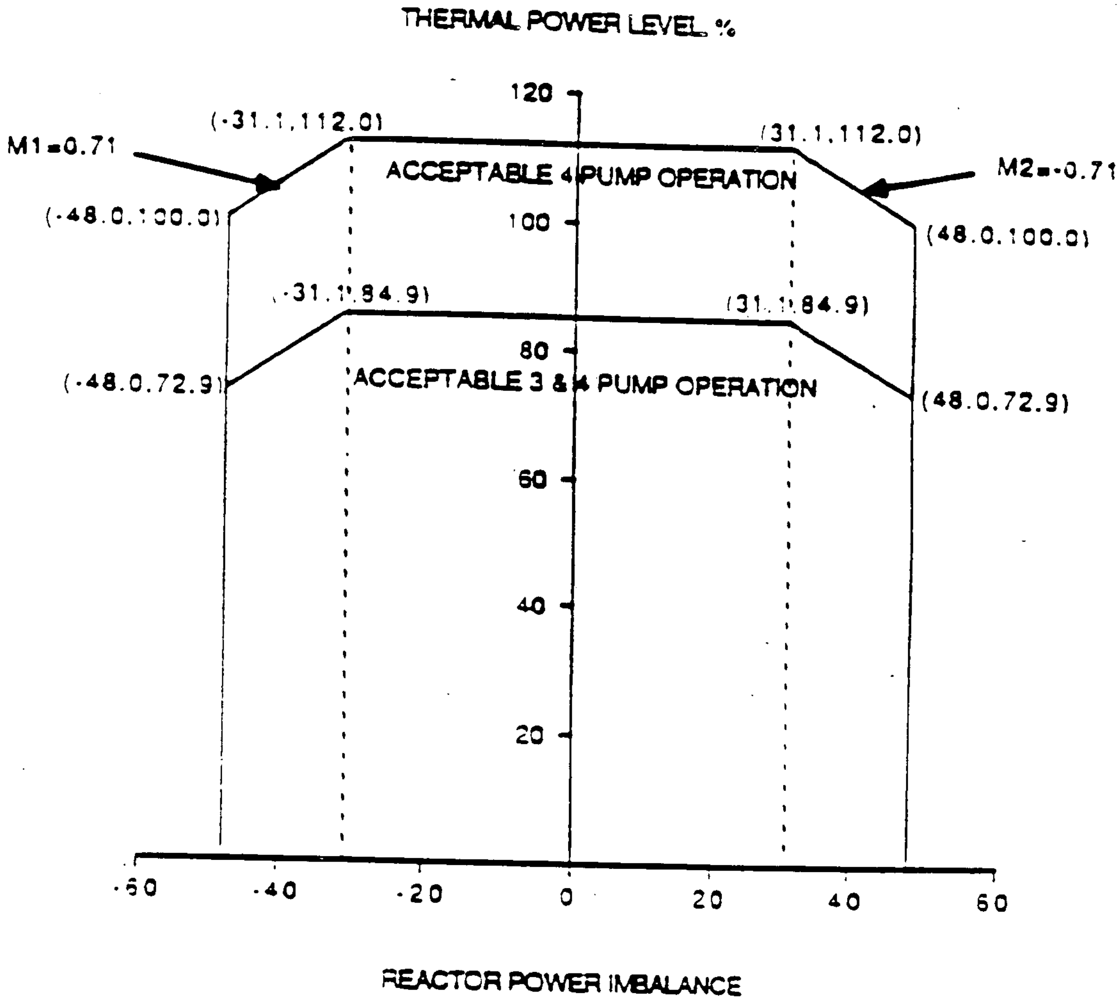


Figure 2.1-2

Axial Power

Imbalance Protective Limits



6.9 CORE OPERATING LIMITS REPORT

Specification

6.9.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle, for the following:

(1) Reactor Protective System Trip Setting Limits for the Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure trip function in Specification 2.3.

(2) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.1.11, 3.5.2.1.b, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.

(3) Quadrant Power Tilt Limits for Specification 3.5.2.4.a, 3.5.2.4.b, 3.5.2.4.d, 3.5.2.4.e, and 3.5.2.4.f.

(4) Power Imbalance Limits for Specification 3.5.2.6

and shall be documented in the CORE OPERATING LIMITS REPORTS.

6.9.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically:

(1) DPC-NE-1002A, Reload Design Methodology II, October 1985.

(2) NFS-1001A, Reload Design Methodology, April 1984.

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989.

6.9.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.