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ADMINISTRATIVE CONTROLS

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CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

46. CENPD-199-P, Rev. 1-P-A, Supplement 2-P-A, "CE Setpoint Methodology," June 1998.
47. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.
48. CEN-396(L)-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2," November 1989 (NRC SER dated October 18, 1991, Letter J.A. Norris (NRC) to J.H. Goldberg (FPL), TAC No. 75947).
49. CENPD-269-P, Rev. 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.
50. CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).
51. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.
52. CENPD-140-A, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," June 1976.
53. CEN-365(L), "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis," June 1988 (NRC SER dated March 13, 1989, Letter J.A. Norris (NRC) to W.F. Conway (FPL), TAC No. 69325).
54. DP-456, F.M. Stern (CE) to E. Case (NRC), dated August 19, 1974, Appendix 6B to CESSAR System 80 PSAR (NRC SER, NUREG-75/112, Docket No. STN 50-470, "NRC SER – Standard Reference System, CESSAR System 80," December 1975).
55. CENPD-387-P-A, Revision 000, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000.

- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

ADMINISTRATIVE CONTROLS (continued)

SPECIAL REPORTS

- 6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

- 6.10.1 The following records shall be retained for at least 5 years:
- a. Records and logs of unit operation covering time interval at each power level.
 - b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
 - c. All REPORTABLE EVENTS.
 - d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.
 - e. Records of changes made to the procedures required by Specification 6.8.1.

(continued on page 6-21)

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation 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 and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through either the CE-1 or ABB-NV correlation. The CE-1 and ABB-NV DNB correlations have been developed to predict the DNB heat 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 the DNB-SAFDL of 1.28 in conjunction with the Extended Statistical Combination of Uncertainties (ESCU). This value is derived through a statistical combination of the system parameter probability distribution functions with the CE-1 or ABB-NV DNB correlation uncertainties. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show conservative loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the DNB-SAFDL is not violated based on the ABB-NV CHF correlation for the family of axial shapes and corresponding radial peaks shown Figure B 2.1-1. The limits in Figure 2-1.1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 107% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The Thermal Margin/Low Pressure and Local Power Density Trip Systems, in conjunction with Limiting Conditions for Operation, the Variable Overpower Trip and the Power Dependent Insertion Limits, assure that the Specified Acceptable Fuel Design Limits on DNB and Fuel Centerline Melt are not exceeded during normal operation and design basis Anticipated Operational Occurrences. Specific verification of the DNB-SAFDL limit using an appropriate DNB correlation ensures that the reactor core safety limit is satisfied.

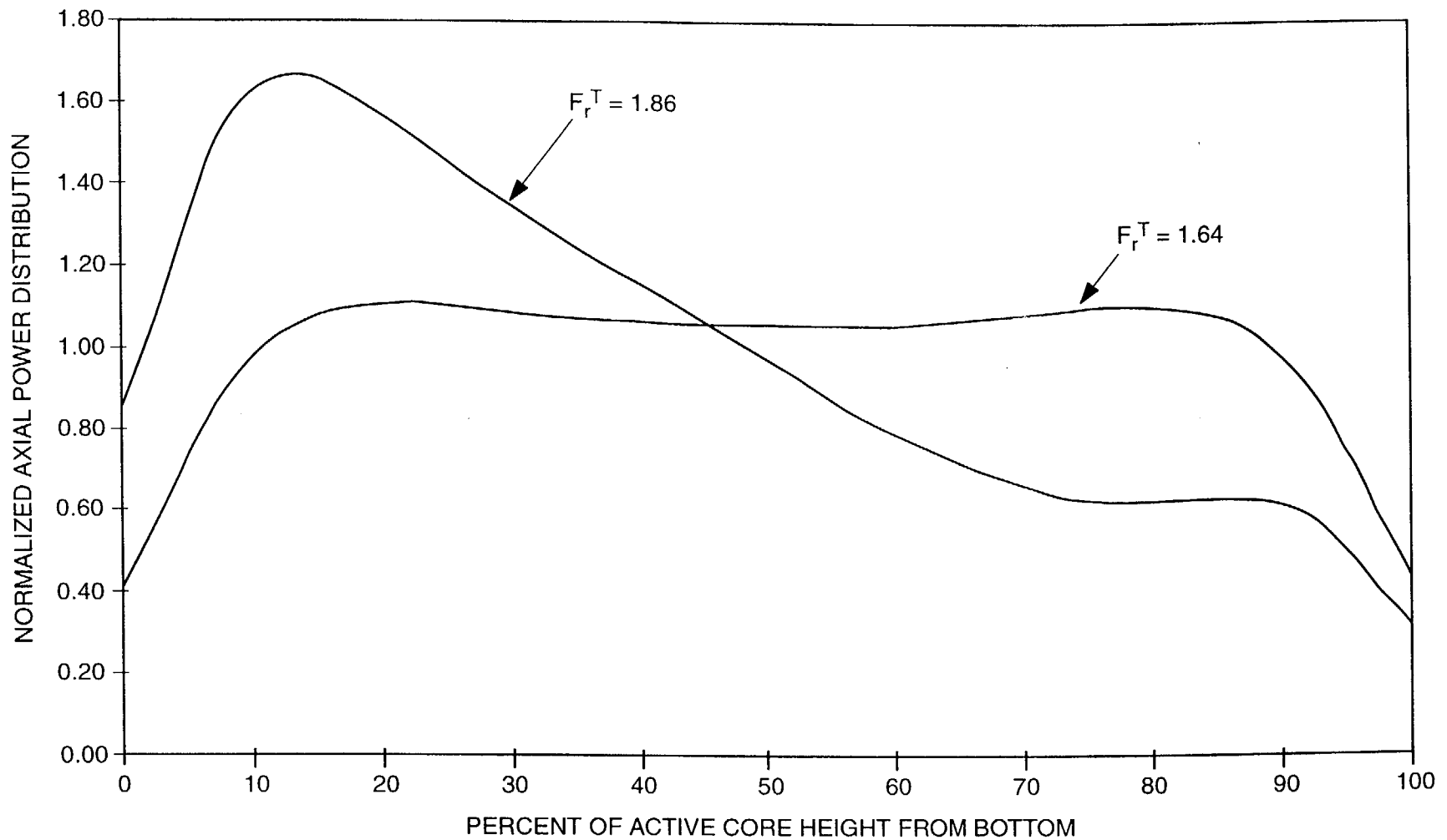


FIGURE B 2.1-1
Axial Power Distributions For Thermal Margin Safety Limits