Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of the curves shown in Figure 15.2.1-1. These curves are based on an $F^N_{\Delta H}$ of 1.58, cosine axial flux shape, and a DNB analysis as described in Section 4.3 of WCAP-8050, "Fuel Densification, Point Beach Nuclear Plant Unit 1 Cycle 2" (including the effects of fuel densification and flattened cladding).

Figure 15.2.1-1 also includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F^N_{\Delta H} = 1.58 \{1 + 0.2 (1-P)\}$ where P is a fraction of rated power when $P < 1.0$. $F^N_{\Delta H} = 1.58$ when $P \geq 1.0$.

The effects of rod bow have been included in the determination of a conservative value for $F^N_{\Delta H}$. Rod bow effects are offset by sources of margin in the design limit DNBR, pitch reduction, design thermal diffusion coefficient and the fuel densification power spike.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of full-length rods that is allowed before the reactor trip setpoints are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occur before reactor trip setpoints are reached. The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30.

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 15.3.10-3 consistent with the Technical Specifications on power distribution control as given in Section 15.3.10 was used in the LOCA analysis. The results of the analyses based on this upper bound envelope indicate a peak clad temperature of less than the 2200°F limit. When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. In the design limit of $F^N_{\Delta H}$, there is eight percent allowance for uncertainties which means that normal operation of the core is expected to result in a design $F^N_{\Delta H} \leq 1.58/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (i.e., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affect F_Q, (b) while the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system. The $F^N_{\Delta H}$ limits in Specification 15.3.10.B.1.a take into account the effects of rod bow. This is further explained in the Basis on page 15.2.1-3.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based upon measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational