TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

December 3, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

On November 9, 1981, we received an informal NRC question concerning hydrodynamic instability at Watts Bar Nuclear Plant. This question was discussed with the NRC project manager and NRR reviewer on December 2, 1981. At that time, we indicated the Watts Bar Final Safety Analysis Report (FSAR) section 4.4 had been revised by Amendment 41 to delete references to the HYDNA program.

Enclosed are FSAR pages 4.4-37,38 and 39 showing the revised information. Figures 4.4-17, -18, -19 and Table 4.4-5 will be deleted from the FSAR because they are no longer referenced. Item 3 of Table 4.1-2 will also be deleted.

If you have any questions concerning this matter, please get in touch with D. P. Ormsby at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

M. R. Wisenburg
Nuclear Engineer

Sworn to and subscribed before me
this 3rd day of Dec. 1981

Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

Enclosure

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The THINC-III Code does not have the capability for evaluating fuel rod thermal response. This is treated by the methods described in Section 15.1.9.

4.4.3.4.2 Fuel Temperatures

As discussed in Section 4.4.2.2, the fuel rod behavior is evaluated utilizing a semi-empirical thermal model which considers in addition to the thermal aspects such items as clad creep, fuel swelling, fission gas release, release of absorbed gases, cladding corrosion and elastic deflection, and helium solubility.

A detailed description of the thermal model can be found in reference [3] with the modifications for time dependent densification given in reference [61].

4.4.3.4.3 Hydrodynamic Instability

The analytical methods used to assess hydraulic instability are discussed in Section 4.4.3.5.

4.4.3.5 Hydrodynamic and Flow Power Coupled Instability

Boiling flows may be susceptible to thermohydrodynamic instabilities[73]. These instabilities are undesirable in reactors since they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components. Therefore, a thermohydraulic design was developed which states that modes of operation under Condition I and II events shall not lead to thermohydrodynamic instabilities.

Two specific types of flow instabilities are considered for W PWR operation. These are the Ledinegg or flow excursion type of static instability and the density wave type of dynamic instability. 41

A Ledinegg instability involves a sudden change in flow rate from one steady state to another. This instability occurs (73) when the slope of the reactor coolant system pressure drop-flow rate curve $\left(\frac{\partial \Delta P}{\partial G}\Big|_{\text{internal}}\right)$ becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve $\left(\frac{\partial \Delta P}{\partial G}\Big|_{\text{external}}\right)$. The criterion for stability is thus $\frac{\partial \Delta P}{\partial G}_{\text{internal}} > \frac{\partial \Delta P}{\partial G}_{\text{external}}$. The W pump head curve has a negative slope ($\partial \Delta P/\partial G/\text{external} < 0$) whereas the reactor coolant system pressure drop-flow curve has a positive slope ($\partial \Delta P/\partial G \mid \text{internal} > 0$) over the Condition I and Condition II operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody[74]. Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single phase

region and causes quality or void perturbations in the two-phase regions which travel up the channel with the flow. The quality and length perturbations in thw two-phase region create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, then the two-phase pressure drop perturbation feeds back to the single phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by Ishii[75] for parallel closed channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability. This method had been used to assess the stability of typical W reactor designs[76,77,78], including Virgil C. Summer, under Condition I and II operation. The results indicate that a large margin to density wave instability exists, e.g., increases on the order of 200 percent of rated reactor power would be required for the predicted inception of this type of instability.

The application of the method of Ishii[75] to W reactor designs is conservative due to the parallel open channel feature of Westinghouse PWR cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high power density. There is also energy transfer from channels of high power density to lower power density channels. This coupling with cooler channels has led to the opinion that an open channel configuration is more stable than the above closed channel analysis under the same boundary conditions. Flow stability tests[79] have been conducted where the closed channel systems were shown to be less stable than when the same channels were cross connected at several locations. The cross connections were such that the resistance to channel to channel cross flow and enthalpy perturbations would be greater than that which would exist in a PWR core which has a relatively low resistance to cross flow. 41

Flow instabilities which have been observed have occurred almost exclusively in closed channel systems operating at low pressures relative to the W PWR operating pressures. Kao, Morgan, and Parker[80] analyzed paralled closed channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Additional evidence that flow instabilities do not adversely affect thermal margin is provided by the data from the rod bundle DNB tests. Many W rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundle.

In summary, it is concluded that thermohydrodynamic instabilities will not occur under Condition I and II modes of operation for W PWR reactor designs. A large power margin, greater than doubling rated power, exists to predicted inception of such instabilities. Analysis has been performed which shows that minor plant to plant differences in W reactor designs such as fuel assembly arrays, core power to flow ratios, fuel assembly length, etc. will not result in gross deterioration of the above power margins. 41

4.4.3.6 Temperature Transient Effects Analysis

Waterlogging damage of a fuel rod could occur as a consequence of a power increase on a rod after water has entered the fuel rod through a clad defect. Water entry will continue until the fuel rod internal pressure is equal to the reactor coolant pressure. A subsequent power increase raises the temperature and, hence, could raise the pressure of the water contained within the fuel rod. The increase in hydrostatic pressure with-in the fuel rod then drives a portion of the water from the fuel rod through the water entry defect. Clad distortion and/ or rupture can occur if the fuel rod internal pressure increase is excessive due to insufficient venting of water to the reactor coolant. This occurs when there is both a rapid increase in the temperature of the water within the fuel rod and a small defect. Zircaloy clad fuel rods which have failed due to waterlogging [81 and 82] indicate that very rapid power transients are required for fuel failure. Normal operational transients are limited to about 40 cal/gm-min. (peak rod) while the Spert tests [81] indicate that 120 to 150 cal/gm is required to rupture the clad even with very short transients (5.5 msec. period). Release of the internal fuel rod pressure is expected to have a minimal effect on the Reactor Coolant System [81] and is not expected to result in failure of additional fuel rods [82]. Ejection of fuel pellet fragments into the coolant stream is not expected [81 and 82]. A clad breech due to waterlogging is thus expected to be similar to any fuel rod