FNP-PAL-081

Offshore Power Systems

P. B. Haga

March 18, 1980

Mr. Robert L. Baer, Chief
Light Water Reactors Branch No. 2
Division of Project Management
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20852

Re: Docket No. STN 50-437; Responses to Requests for Additional Core Thermohydraulic Information

Dear Mr. Baer:

Attached is the additional information requested in the attachment to your letter dated October 17, 1979. This information will be added to Appendix B of the Plant Design Report at the next amendment.

Very truly yours,

P.B. Haga

P. B. Haga

/lel

Attachment

CC: A. R. Collier

B001
SE
1/1

ADD:	Ltr	Enc
R. Denise	1	1
F. Schroeder	1	1
L. Phillips	1	1
M. McCoy	1	1

8003280531

RESPONSE TO NRC REQUESTS FOR

ADDITIONAL INFORMATION

DATED

OCTOBER 17, 1979

221.1 (4.4) Provide the radial pressure gradient in the upper and lower plenums and at the core inlet and outlet for steady state and transient conditions for each allowable loop configuration. Provide an explanation of how the radial pressure gradients are included in the thermal-hydraulic design calculations. Discuss and support by calculations, the differences in hot channel pressure drop, flow, enthalpy rise and minimum DNBR relative to the assumption of a uniform pressure at the core boundaries.

Response

DNB analyses are based on uniform inlet velocity and exit pressure distributions. Data from several 1/7 scale model tests and THINC analyses of various inlet flow distributions have led to a conservative design basis of 5% reduction in flow to the hot assembly. Section 5.6 of Reference 1 presents analyses which verify the adequacy of the design assumption of a uniform exit pressure distribution.

The effect of core outlet radial pressure gradients on DNB analysis has been shown to be negligible in four-loop 193 assembly cores. An analysis was performed which assumes a cosine upper plenum radial pressure gradient with a maximum value of 5 psi at the core center and 0 psi at the core periphery for four-loop and three-loop operation. The results of these analyses showed that there was no effect on the minimum DNBR (to three significant figures) of this radial pressure gradient on four-loop or three-loop operation.

In performing this analysis the hot assembly was assumed to be in the center of the core where the greatest flow reduction near the core outlet will occur due to the radial pressure gradient. In addition, an axial power distribution extremely peaked to the top of the core (+30% axial offset was assumed. This axial power distribution is more severe than would be expected during plant operation.

Thus, the use of a uniform upper plenum pressure distribution in thermal-hydraulic design is acceptable.

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(1) Hochreiter, L.E., and H. Chelemer, "Application of te THINC-IV Program to PWR Design", WCAP-8504 (Proprietary) and WCAP-8155, September 1973.

221.2 Provide a description of how the effects on the core flow and
(4.4) pressure drop of possible crud deposits are included in the thermal-hydraulic design.

Provide a description of the instrumentation available which would alert the reactor operator to an abnormal core flow or core pressure drop during steady-state operation.

Response

Operating experience to date has indicated that a flow resistance allowance for possible crud deposition is not required. There has been no detectable long-term flow reduction reported at any plant. Inspection of the inside surfaces of steam generator tubes removed from operating plants has confirmed that there is no significant surface deposition that would affect system flow. Although all of the coolant piping surfaces have not been inspected, the small piping friction contribution to the total system resistance and the lack of significant deposition on piping near steam generator nozzles support the conclusion that an allowance for piping deposition is not necessary. The effect of crud enters into the calculation of core pressure drop through the fuel rod frictional component by use of a surface roughness factor. Present analyses utilize a surface roughness value which is a factor of three greater than the best estimate obtained from crud measurements from several operating Westinghouse reactors.

Instrumentation available to alert the operator to abnormal core flow or core pressure drop is as follows:

1. Primary flow indication is provided by the RCS flow meters. There are 3/loop and read from 0-100%. Any significant flow reduction would appear on these meters.

2. There are several methods that could be used to infer flow. They are:

   a. With rods in the automatic control mode, reduced flow would result in lower core power as rods drove in attempting to maintain Tavg.

b. With rods in the manual control mode, reduced flow would result in higher Tavg.

c. RCP amps reading higher or lower than normal, could indicate abnormal flow (pump or motor malfunction primarily).

d. Significant flow reductions in a particular core quadrant could be indicated by a power mismatch between the various power range detectors.

e. Sustained local flow stoppages could be detected by incore flux maps and core exit thermocouples.

3. There are alarms that would alert the operator to low RCS flow as indicated by the RCS flow meters, high RCS temperatures, abnormal RCP and motor temperatures and RCP trip.

4. Reactor trips are generated by low RCS flow, low RCP bus voltage and frequency, and high temperature.

221.3 (4.4) Provide a commitment to address the following aspects of rods bowing in the FSAR for plants referencing FNP (1-8):

1. to fully define the gap closure rate for prototypical bundles;

2. to determine by appropriate experiments the DNB effect that bounds the effect of gap closure;

3. to include the effect of rod bowing in the final design and safety analysis calculations.

Response

DNB analyses (which will be reported during the final FNP design approval phase) will be performed such that generic DNBR margins described in the "Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors (Revision 1) February 16, 1977" will be available for offsetting rod bow penalties. The appropriate rod bow penalty and any operating restriction in the technical specifications, if required, will be addressed prior to the issuance of an Operating License to the owner of the first FNP.

221.4 (4.4) Floating Nuclear Plants (1-8) used the HYDNA code to describe the effect of open channel flow on thermal-hydraulic flow instability. Information supplied by Westinghouse has been insufficient to support a conclusion that the HYDNA code conservatively predicts the onset of flow instability in the core. To support such a conclusion, either (1) provide a complete description of the HYDNA code and its use in the analysis, or (2) provide a discussion excluding the HYDNA code which supports the contention that the core is thermal-hydraulically stable.

Response:

Boiling flows may be susceptible to thermohydrodynamic instabilites.[1] 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 such that operation under Condition 1 and 11 events does not lead to thermohydrodynamic instabilities.

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

A Ledinegg instability involves a sudden change in flow rate from one steady state to another. This instability occurs[1] when the slope of the reactor coolant system pressure drop-flow rate curve ($\frac{\partial \Delta P}{\partial G}\Big|_{\text{INTERNAL}}$) becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve ($\frac{\partial \Delta P}{\partial G}\Big|_{\text{EXTERNAL}}$). The criterion for stability is thus $\frac{\partial \Delta P}{\partial G}\Big|_{\text{INT.}} > \frac{\partial \Delta P}{\partial G}\Big|_{\text{EXT.}}$. The Westinghouse pump head curve has a negative slope ($\frac{\partial \Delta P}{\partial G}\Big|_{\text{EXT}} < 0$) whereas the reactor coolant system pressure drop-flow curve has a positive slope ($\frac{\partial \Delta P}{\partial G}\Big|_{\text{INT}} > 0$) 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.[2] 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 the 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, 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[3] 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 Westinghouse reactor designs [4,5,6] 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% of rated reactor power would be required for the predicted inception of this type of instability.

The application of the method of Ishii[3] to Westinghouse 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[7] 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.

Flow instabilities which have been observed have occurred almost exclusively in closed channel systems operating at low pressures relative to the Westinghouse PWR operating pressures. Kao, Morgan and Parker[8] analyzed parallel 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 Westinghouse rod bundles have been tested over wide ranges of operating conditions wlth 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 Westinghouse 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 Westinghouse 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.

References:

1. J. A. Boure, A. E. Bergles, and L. S. Tong, "Review of Two-Phase Flow Instability," Nucl. Engr. Design 25 (1973), pp. 165-192.

2. R. T. Lahey and F. J. Moody, "The Thermal Hydraulics of a Boiling Water Reactor," American Nuclear Society, 1977.

3. P. Saha, M. Ishii, and N. Zuber, "An Experimental Investigation of the Thermally Induced Flow Oscillations in Two-Phase Systems," J. of Heat Transfer, Nov. 1976, pp. 616-622.

4. V gin C. Summer FSAR, Docket #50-395.

5. Byron/Braidwood FSAR, Docket #50-456.

6. South Texas FSAR, Docket #50-498.

7. S. Kakac, T. N. Veziroglu, K. Akyuzlu, O. Berkni, "Sustained and Transient Boiling Flow Instabilities in a Cross-Connected Four-Parallel Channel Upflow System," Proc. of 5th International Heat Transfer Conference, Tokyo, Sept. 3-7, 1974.

8. H. S. Kao, C. D. Morgan, and W. B. Parker, "Prediction of Flow Oscillation in Reactor Core Channel," Trans. ANS, Vol. 16, 1973, pp. 212-213.