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Mr. Robert L. Tedesco
Assistant Director of Licensing
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Response to Request for
Additional Information -
Core Thermal Hydraulics

Dear Mr. Tedesco:

In a letter dated October 1, 1981, we provided responses to your concerns in the area of core thermal hydraulics. Subsequent review of our responses by your reviewer resulted in the need for additional clarifications. This letter forwards our responses to these items. It is our intention to incorporate these responses in a subsequent amendment to our Final Safety Analysis Report.

Very Truly Yours,

Dalwyn R. Davidson

Dalwyn R. Davidson
Vice President
System Engineering and Construction

DRD: mlb

Attachment

cc: M. D. Houston
G. Charnoff, Esq.
NRC Resident Inspector

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492.3 What are the values of the following for Perry?

(Table 4.4-1)

a) Design basis maximum core support plate pressure drop(normal + upset);

b) Design basis maximum allowable channel wall pressure drop.

Response

The design basis core support plate pressure drop(normal + upset) is 26 psid; the maximum calculated channel wall pressure drop at normal conditions is 15.6 psid. The design basis for BWR fuel channels and vessel internal structures (including the core support plate) with respect to pressure differentials is that the component in question maintain structural integrity when subjected to certain load combinations. A pressure difference acting on a given component is merely considered as one of the loads in those combinations. As such, it is not necessarily meaningful to specify a design basis maximum pressure drop without also specifying the other associated loads. Furthermore, it should be made clear that although a given maximum pressure drop may be used in such calculation that does not imply that the component in question could not withstand an ever greater differential pressure.

492.4
(4.4.2.6)

Have the core plate pressure drop measurements ever been done for the operating BWRs with 8x8R or P8x8R fuel with two water rods? If not, do you intend to do these measurements?

Response

We are not aware of any operating plants with 100 percent 8x8R or P8x8R fuel with two water rods to provide such a comparison of calculated with measured core plate pressure drop. The purpose of the comparison in Table 4.4-7 is to verify the thermal hydraulic calculation. As the methods of analyzing those cores with 8x8R or P8x8R fuel are no different than those for 7x7 or 8x8 fuels, it is anticipated that any such comparison would yield results nearly identical to those given in Section 4.4.2.6 and in Table 4.4-7. Furthermore, the plant operator has the capability, to monitor the core plate pressure drop at any time during operation.

492.9 (4.4.4.5.2) What fraction of the fuel bundle is "water rod flow"? Did you verify your calculations with previous measurement data?

Response

The nominal water rod flow fraction for Perry at rated conditions is 1.35% of the total core flow; models used in the calculations of this fraction were derived from experimental data. The calculation of this fraction was accomplished by the computer program ISCØR which used the same steady state thermal hydraulic mathematical module described in NEDO-20953A. Also, see response to question 492.11.

492.11 (4.4.4.5)	You have not cited the name, version, or reference of the computer program used in this sub-section.

Letter from N. W. Curtis (Pennsylvania Power and Light Company) to B. J. Youngblood (NRC), "Response to NRC question on Susquehanna FSAR," dated March 25, 1981, states that name of the computer program is "ISCØR" and reference is "General Electric Document NEDO-20953, May 1976, Chapter 4."

Please confirm ISCØR has been used for Perry.

What version number of ISCØR is the latest version? Has this version been applied to Perry? If the reference of this version is different from GE Document NEDO-20953, provide the document or the reference. Also describe any significant changes of this version of ISCØR code over the previous version of ISCØR.

Response

The computer program cited in Section 4.4.4.5 is named ISCØR. The ISCØR computer program and another GE program PANACEA (3 dimensional BWR core simulator) use the same steady state thermal hydraulic mathematical module described in NEDO-20953-A dated January 1977. The program ISCØR and the calculations used for Perry are consistent with the technical content of NEDO-20953-A dated January 1977.

The details of ISCØR and its associated proprietary documentation are available for review at GE in San Jose, however, the following provides a general description of the code.

A. Summary

The purpose of ISCØR is to perform a steady state thermal hydraulic analysis of a nuclear reactor core consisting of parallel but noncommunicating flow channels. The code also allows evaluation of the critical power based on fluid properties within each channel by the GEXL critical power correlation. Input to the code includes core geometry, operating pressure level, and distribution within the core, operating pressure level, and core inlet enthalpy. The code calculates the required total flow and flow distribution for input core pressure drop or the flow distribution and core pressure drop for an input total core flow. The code considers the pressure drop and flow in the core of the reactor only. Detailed modeling of the bypass region, leakage flow paths, and water rod hydraulics is included. Thermal performance calculations are carried out using the GEXL critical quality-boiling length correlation. Output includes core pressure drop for a required core flow or core flow for a required core pressure drop, core flow distribution among the channel types, and the critical power ratio in each of the channel types. The code is restricted to the use of light water as the coolant.

B. Core Hydraulic Model

The hydraulic analysis assumes that the total core flow rate is divided among the fuel assemblies such that each assembly experiences the same pressure drop from the lower to the upper plenum. The total core flow rate is divided into in-channel flow and bypass or leakage flow. The in-channel flow may be further divided into active flow and flow passing through the interior of the water rods or rods which do not contain fuel.

The solution is limited to twenty-four distinct fuel assembly types which are treated as parallel flow paths. To analyze an entire core, it is necessary to group the fuel bundles into twenty-four or fewer fuel assembly types. All of the fuel bundles belonging to a specific fuel assembly type are assumed to have identical hydraulic characteristics, power level, and power shape.

The core hydraulic model uses a single core reference pressure for the evaluation of all coolant physical properties. This assumption has a negligible impact on the results at normal operating pressure, but could become more significant at low pressures.

C. Core Pressure Drop

The core or plenum-to-plenum pressure drop is calculated by summing the friction, elevation, local, and acceleration pressure drop components. Frictional and elevation losses are summed over three separate regions. First, a heated core section with axially constant geometry. Second, a unheated section (representative of the fuel rod gas plenum) with the same geometrical cross-section as the heated region. Third, an unrodded section (representative of the channel above the heated bundle) with the same geometrical cross-section as the channel.

D. Core Power Distribution

The core power distributions are presumed known and are supplied as input to the ISCØR code. Current physics codes such as the BWR three-dimensional simulator which is used to calculate reactor power distributions include the same void and subcooled boiling models used in ISCØR but they cannot perform the detailed pressure drop calculations included in ISCØR. If power distributions used by ISCØR are obtained from the physics design code, the power distributions and void distributions should be compatible.

Each fuel assembly type is characterized by a normalized axial power profile, a normalized radial power ratio (the ratio of the bundle power to the average bundle power), and a normalized local peaking profile. The local peaking profile is only used in calculating the local maximum heat flux or linear heat generation rate and is not used in the coolant enthalpy rise calculations.

The core power input by the user is divided into two parts: in-channel coolant power and a bypass region power. The bypass flow is heated by neutron-slowing down and gamma heating in the water and by heat transfer through the channel walls. Heat is also transferred to the bypass flow from structural and control elements which are themselves heated by gamma absorption and by , a reaction in the control material. This fraction of the total reactor power that goes into the bypass flow is called the reactor bypass power fraction. The remainder of the reactor power is the in-channel coolant power. The in-channel power is divided into the active power which heats the active coolant and the water rod power which heats the water rod coolant stream.

The heated length of each fuel assembly type can be divided into as many as fifty equal length nodes. The normalized axial power factor for each node is input and is the ratio of the average power in the node to the average nodal power in the fuel assembly. The heated length of individual fuel assemblies can vary, but the number of heated nodes must be constant for all fuel assembly types. Thus, if the heated length varies between different fuel assemblies, the nodal length will also differ from one fuel assembly to another.

E. Core Coolant Enthalpy Distribution

The coolant enthalpy distribution in each channel type is calculated using a simple heat balance on each axial node. The enthalpy rise calculated is a thermodynamic equilibrium enthalpy for the channel and is based on average node power.

Since the power is assumed to be uniform over each axial node, the mid-node enthalpy is just the average of the end-of-node value and the end-of-node enthalpy of the previous node.

Similar calculations are carried out for the bypass coolant and water rod coolant enthalpy distributions.

F. Core Critical Power Calculations

The critical power calculations are not carried out until the pressure drop calculations have been completed and the channel flow and fluid conditions are known. The critical power of each fuel assembly type is calculated for fixed values of the inlet enthalpy and mass flux by adjusting the bundle power until the minimum thermal margin in the bundle approaches one. The bundle critical power ratio is defined as the ratio of the critical power to the actual bundle power.

The critical quality is a function of the bundle mass flux, the system pressure, heated length, the bundle thermal diameter (a quantity which is defined as four times the bundle flow area divided by the perimeter of all rods in the bundle including the water rods), the bundle R-factor (a measure of the local peaking pattern), and the distance from the bulk boiling boundary (known as the boiling length).

The core minimum critical power ratio is the minimum value of the critical power ratios for all of the fuel assembly types in the core.

492.16 (4.4) You have not cited the name, version, and reference of the core wide transient analysis code (i.e., ODYN or REDY) and for the GETAB-MCPR evaluation of the transients. Please provide name, version, and reference of these two codes used for Perry.

Response

The REDY code, as documented in NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," was used for the core wide transient analysis as shown in Chapter 15. Limiting pressurization events evaluated with the ODYN code will be provided in the near future. All the GETAB-MCPR evaluation of the transients was performed with SCAT code as documented in NEDO-20566, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in accordance with 10CFR50, Appendix K." However, in order to make SCAT more compatible to ODYN output a modified version of SCAT has been prepared in conjunction with ODYN. The NRC was notified of this modified version of SCAT in a letter from GE's K. W. Cook to F. Schloeder and D. Eisenhut (MFN - 171 - 79) dated July 20, 1979.

492.19 (4.4.4.6) The staff is performing a generic study of the hydrodynamic stability characteristics of LWRs under normal operation, anticipated transients, and accident conditions. The results of this study will be applied to the staff review and acceptance of stability analyses and analytical methods now in use by the reactor vendors. In the interim, the staff concludes that past operating experience, stability tests, and the inherent thermal-hydraulic characteristics of LWRs provide a basis for accepting the Perry stability evaluation for normal operation and anticipated transient events. However, in order to provide additional margin to stability limits, natural circulation operation of Perry will be prohibited until the staff review of these conditions is complete. Any action resulting from the staff study will be applied to Perry.

Response

It is believed that the stability characteristics for Perry meet the ultimate stability limit even with the natural circulation operation. Additionally, CEI is a member of (LRG-II) the Licensing Review Group which is pursuing a common resolution to this issue (11-CPB).

492.22 No analysis has been presented for MCPR limits or stability characteristics for one loop operation. One loop operation will not be permitted until supporting analyses are provided and are approved by the staff.

Response

Stability and transient analysis for operating CPR limits presented in the FSAR are for two loop operation. Natural circulation conditions are the same in either case. If CEI decides to operate the plant with one recirculation loop, additional support analyses will be provided and submitted to the NRC for approval.

492.24 (4.4) How do you adjust operating limit MCPR values for the operation at lower than 100 percent power and 100 percent flow conditions?

Response

The operating limit minimum critical power ratio (MCPR) at off-rated operating states is determined from the $MCPR_f$ and $MCPR_p$ curves which are functions of core flow and power, respectively. These curves and the associated analytical for these curves are integrated into the plant technical specifications; as such, further information regarding the off-rated operating limit MCPR will be provided at the time of Perry tech specs submittal.

492.26 (4.4.6) Provide an evaluation of you LPMS for compliance with Regulatory Guide 1.133. Justify any deviations.

Response

The response to this question is provided in revised Section 4.4.6.1.

The PNPP LPMS design conforms to guidelines of Regulatory Guide 1.133, Rev. 1, as identified in Table 1.8-1 of the FSAR.