ATTACHMENT

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

RELOAD DESIGN METHODOLOGY
TECHNICAL REPORT

(NFS - 1001)

Revision 3

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Technical Report

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6.4 Core Inlet Conditions

The first block of the Figure 6.1, Core Inlet Conditions, represents the hand calculations required to determine the core mass flow rate and the core inlet temperature (enthalpy) for each plant operating condition to be analyzed.

The reactor coolant pumps are constant volumetric flow pumps; thus the RCS mass flow rate varies with cold leg temperature. Further, the integrated control system maintains a constant average temperature over the power range of 15-100 percent, which requires that cold leg temperature decrease with increasing core power. These two factors when combined require that an iterative calculation be used to determine core inlet temperature and mass flow rate over the power range analyzed. Additional density changes (flow corrections) are made to account for parametric variations in the core inlet temperature and outlet pressure as well as for the temperature and pressure errors which are applied during the analysis.

6.5 Reference Design DNBR Analysis

This section represents block number 2 of Figure 6.1 and discusses the method used to determine the reference design DNBR, which is reported in Table 6-1, Thermal-Hydraulic Design Conditions, of each Reload Report.

A two stage analysis is used to determine this reference DNBR: 1) a core-wide analysis and 2) a hot assembly/hot channel analysis. These two stages are described in subsections 6.5.1 and 6.5.2, which follow.

6.5.1 Core-Wide Analysis

The CHATA[10] (Core Hydraulic And Thermal Analysis) program is used to determine the core-wide flow distribution. CHATA is a closed channel model (no energy or mass interchange among assemblies) that varies flow to each assembly until 3
each one has the same pressure drop and the sum of the assembly flows is equal to the input core flow.

Total core flow effective for heat transfer is input into CHATA, which models single fuel assemblies on an eighth-core symmetric basis to determine the core flow distribution. The calculated result of particular importance that is derived from this CHATA core flow distribution is the hot assembly flow, which is subsequently input into the hot assembly analysis described in Subsection 6.5.2.

Primary inputs to this core-wide analysis are core flow effective for heat transfer, individual fuel assembly geometries, form loss coefficients, the generic radial peaking distribution, the 1.5 design cosine axial flux shape, and core operating conditions.

Core flow rate is one of the most important inputs to the thermal-hydraulic analyses, and the possibility exists that this input flow can change depending on measured flow. Reloads are now being designed based on 106.5% of the original design reactor coolant system flow rate of 88,000 gpm per pump. The 106.5% figure was selected based on the lowest measured flow rate less measurement uncertainties.

Core flow is equal to total loop flow less the bypass flow, which is defined as that part of the flow that does not contact the active heat transfer surface area. These bypass flow paths are 1) core shroud, 2) core barrel annulus, 3) control rod guide tubes and instrument tubes, and 4) all interfaces separating the inlet and outlet regions. A typical value of the design bypass flow is 8.10%; however, this flow rate is dependent on the number of orifice rod and burnable poison rod assemblies. Removal of orifice rod assemblies and/or changes in the number of burnable poison rod assemblies and retainers cause both changes in the core bypass flow rate and in the core flow distribution. Such changes will either be reflected appropriately in the core-wide flow distribution or will be conservatively enveloped.

The basic assumption for the core inlet flow distribution, which is based on vessel model flow tests and Oconee 1 core pressure drop measurements, is that the isothermal flow distribution is relatively flat with a maximum deviation of

less than 5% for 4-pump flow conditions. Therefore, the hot assembly is assumed to receive only 95% of the nominal assembly flow for this isothermal, four pump condition.

These isothermal flow maldistribution factors are considered during the core-wide CHATA analysis by the use of an additional form loss coefficient located at the entrance of the "hot" assembly. However, it is important to note that the numerical value of this form loss coefficient is determined in an isothermal flow distribution analysis to be consistent with the fact that the vessel model flow test is an isothermal model. Subsequently, when the CHATA core-wide model is run at power conditions, a significantly larger hot assembly flow maldistribution results due to the radial peaking factor of the hot assembly. Further conservatisms imposed on the hot assembly during the core-wide analysis are minimum spacing effects on the flow area and on the form loss coefficients.

6.5.2 Hot Assembly/Hot Channel Analysis

The second step toward determining the reference design DNBR is the hot assembly/hot channel analysis, which is also represented by block number 2 of Figure 6.1. The term "hot" assembly refers to that fuel assembly with the highest radial peaking factor(actually the intersection of four 1/4 assemblies). The term hot channel refers to the subchannel with the highest single pin peaking factor. This subchannel occurs within the hot assembly and is generally formed by the intersection of four fuel pins, although the hot channel could occur in a pin-control rod subchannel. (Hot channel factors are always included in all subchannel types within the hot bundle to permit this possibility.)

The conservative hot assembly flow rate from Section 6.5.1 is input into the TEMP[14] (Thermal Energy Mixing Program) code for detailed analysis of the single "hot" assembly. The Oconee hot assembly pin by pin peaking distribution conservatively models the intersection of the pin arrays of four 1/4 assemblies, with a relatively flat local peaking gradient, to conservatively minimize beneficial energy mixing effects. This generic pin-by-pin power distribution includes the design radial peaking factor of 1.714 at the hot channel. All hot 3

channel factors are applied, so that the resulting DNBR calculated represents the worst case (lowest) DNBR for the reactor core for the specified input conditions. The assembly-wide modeling utilizes an open channel calculation; that is, it calculates energy interchange between channels at each calculational increment up the channel. This energy interchange reduces the enthalpy rise in the hot channel, thereby raising the minimum DNBR and permitting a higher allowable peaking factor for the reactor core for conditions when DNBR is limiting. However, the model does not include mass interchange between subchannels.

The outputs of primary importance from the hot assembly/hot channel analysis are the hot channel minimum DNBR and the hot channel flow rate. The hot channel minimum DNBR from the 112% overpower analysis is the reference design DNBR. In general, these outputs of minimum DNBR and hot channel flow are used to establish the equivalent hot channel model of Section 6.6, which itself is used for parametric studies.

6.5.3 Hot Channel Factors

The following hot channel factors are utilized in the thermal-hydraulic analyses of Sections 6.5, 6.6, 6.7, and 6.9. Additional hot channel factors are included in the analyses of Section 6.8 and are described therein.

The flow area reduction factor, is 0.98 on the hot unit and the hot pin-rod cells, and is 0.97 on the instrument guide tube, wall and corner cells. This factor, a statistical computation from measured as-built rod gaps, is applied to the channel flow area at each increment.

The hot channel factor on average pin power, is 1.0107. This factor accounts for variations in average pin power caused by differences in the absolute number of grams of U-235 per rod. The loading tolerance on U-235 per fuel stack and variation on the powder lot mean enrichment are considered in determining the factor.

The hot channel factor due to manufacturing tolerances, is 1.014. Variations in pellet density, pellet cross-sectional area, weight per unit length, local enrichment, and local outer clad diameter are all accounted for in this factor.

pressure drop regardless of variations in local peaking and axial power shape. In other words, hot channel flow rate will be adjusted by the code to satisfy core-wide pressure drop as local conditions are varied. The axial power shapes input to these parametric hot channel runs are smooth cosine curves whose peak can be specified at various distances up the channel for each series of axial peaking factors. To obtain the maximum allowable peaking factor for each data point, power input to the channel is increased until the limiting DNBR of 1.4326 is reached. This process determines a maximum allowable total peak for a specified axial peak and its location.

After completion of these parametric analyses, two sets of generic DNBR curves or Maximum Allowable Peaking (MAP) curves are determined. One set is used for DNB operational offset limits, and the second set is used for RPS DNB offset limits. The generic DNBR curves used as operational limits are a conservative overlay of 1) the generic DNBR curves used for RPS offset limits, and 2) another set of MAP curves which have the reference design DNBR as their basis. Both sets of limits consider the extremities of the P-T core protection envelope (619°F and 1800 psig) as potential core operating conditions. Thus both the operational DNB offset limits and the RPS DNB offset limits have considered the worst case temperature and pressure envelope permitted by the RPS.

The last step in the thermal-hydraulic analysis is to take actual power shapes that gave the lowest DNBRs during the maneuvering analysis and input these irregularly shaped axial curves into the hot channel code to verify conservatism of the corresponding cosine curves used to develop the generic DNBR curves. A typical set of generic DNB curves is provided in Figure 6.3.

6.8.3 Hot Channel Factors

The following additional hot channel factors on local heat flux are utilized in the thermal-hydraulic analyses for developing the generic DNBR curves:

1.026 = penalty incurred to increase calculated axial powers since flux depressions at the spacer grids are ignored.

1.024 = the ratio of the total nuclear uncertainty of 1.075 to the radial nuclear uncertainty of 1.05.

Thus, in determining the generic DNB curves, the normal value of Fq" is increased from 1.014 to 1.065.

6.9 Transient Analysis - Determination of the Flux - Flow Ratio

During a loss of one or more reactor coolant pumps, the core is prevented from violating the 1.4326 minumum DNBR criterion by a reactor trip that is initiated by exceeding the allowable reactor power to reactor coolant flow ratio setpoint. Loss of one or more reactor coolant (RC) pumps is also detected by the RC pump monitors. That is, independently of the power to flow trip, loss of one RC pump will result in an automatic reactor runback. Similarly, loss of two or more RC pumps from above 55% full power will cause a reactor trip.

The thermal-hydraulic analysis that is used to set the power to flow trip setpoint for coastdown protection conservatively assumes the loss of two RC pumps. The transient is analyzed using the RADAR[15] code to assure that the 1.4326 mini- 3
mum DNBR criterion is not violated at anytime during the loss of one or more RC pumps.

The steady state thermal-hydraulic analysis provides the starting point for the transient analysis. The power to flow setpoint itself is derived from this analysis by varying the time of reactor trip following the loss of two RC pumps (that is by considering various trip setpoints) until the minimum ratio required to maintain the minimum DNBR of 1.4326 has been determined. Calculation of the actual (error corrected) power to flow setpoint used at the nuclear station is described in Section 7.3.2.

6.10 Application of the Rod Bow Penalty

In existing thermal-hydraulic analyses, a very conservative DNBR penalty is included to account for rod bowing effects. This penalty (11.2%), however, has been reduced by 1% because of the flow area (rod pitch) reduction factor already included in the thermal hydraulic analysis.

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For some reloads, additional credit can be applied based on the fact that primary coolant flow can be proven to be higher than the 106.5% design flow.

10. REFERENCES

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EPRI-SHUFFLE

The EPRI-SHUFFLE program will read a PDQØ7 concentration file, make certain modifications to this file, and write a new updated concentration file. This procedure is accomplished by defining "assembly regions" in the program input. Assembly regions are square arrays of mesh points containing depletable nuclide concentrations and superimposed on the original PDQØ7 geometry. These assembly regions are then used to describe the movement of existing nuclide concentrations by translation, reflection and/or rotation. In addition, new fuel concentrations can replace spent fuel concentrations in selected assembly regions described in the program's input.

EPRI-SUPERLINK

SUPERLINK accesses data on the files produced by EPRI-FIT and together with relevant input information for file management and for data processing control produces polynomial coefficients for use in EPRI-NODE.

PDQØ7

See EPRI-PDQØ7 Modifications.

NODE UTILITY CODE (NUC)

The NUC program is a package of subroutines that performs any necessary utility function to EPRI-NODE files. The major subroutines are:

I. FILE - this mode lists, merges, purges, adds, rearranges, edits, etc. the NODE cases on one or more history files.

II. FLEX - this mode takes an existing file, expands or collapses it to a new problem size, and then stores it on a new disk.

III. COPY - this mode copies a given history file from disk storage (working file) to magnetic tape storage (permanent backup file) and vice versa.

IV. MARGINS - this mode performs those operations which are necessary to calculate CFM, DNB, and LOCA margins from an input history file(s). It also plots the results in the form of a "fly speck" graph.

TACO2

TACO2 conservatively predicts fuel pin temperature and fuel pin pressure. It includes models for fuel densification, fuel swelling, fuel restructuring, gas release, cladding creep, and gap closure. 3

TEMP

TEMP is a steady state open channel thermal hydraulic code that considers energy mixing between channels and is used to calculate flow distribution among individual channels in an assembly or a cluster of fuel pins. It calculates flow, pressure drop, coolant parameters up the channel, and DNBR.

RADAR

RADAR performs a thermal analysis of a slow reactor transient such as the loss of a primary pump, computing as a function of time fuel pin and clad surface temperatures, DNBR, and coolant thermodynamic conditions when given pin power and either channel flow or pressure drop as a function of time.

Appendix B

List of Revisions

Revision	Date Issued
1	May 1980
2	January 1981
3	April 1981