UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THERMAL HYDRAULIC METHODS USING THE FIBWR CODE

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325/324

1.0 INTRODUCTION

By letter dated May 31, 1983 (Ref. 1), Carolina Power & Light Company (CP&L) submitted NF-1583.04, "Verification of CP&L Reference BWR Thermal-Hydraulic Methods using the FIBWR Code" (Ref. 2), for staff review. The topical report provides a description of CP&L's modeling methods and their benchmark results for applications of the FIBWR code for thermal hydraulic analyses for the Brunswick Plant Units 1 and 2.

FIBWR is a steady state thermal hydraulic code developed by Yankee Atomic Electric Company (Ref. 3, 4, 5). It determines the flow and void distributions within a boiling water reactor (BWR) core by solving the steady-state one-dimensional equations of continuity, momentum and energy. FIBWR computes the coolant mass flow rate in each channel and in the bypass region for either a given total core mass flow or a specified total core pressure drop. A boiling length-critical quality critical heat flux (CHF) correlation, GEXL, is added to the FIBWR code by CP&L for the critical power ratio (CPR) calculations. The FIBWR code had previously been reviewed and approved by the staff (Ref. 6) for application to Vermont Yankee Reloads. Therefore, this review is to investigate the validity and applicability of the FIBWR code for the reload thermal hydraulic analysis by CP&L for the Brunswick BWRs.

2.0 STAFF EVALUATION

The evaluation of the topical report consists of review of FIBWR code validation, modeling methods, benchmarking and application of FIBWR for Brunswick BWR core thermal hydraulic analysis. The evaluation follows:

2.1 FIBWR Code Validation

Since the original FIBWR code developed by YAEC had previously been reviewed and accepted by NRC, this review is concentrated on the changes made by CP&L to the original FIBWR codes. CP&L has added the GEXL CHF correlation to the code for steady state CPR calculations. The GEXL correlation as described in General Electric report NEDE-25422, (Ref. 7) has been approved by NRC for application to GE fuels in BWR core within its ranges of applicability. The validation of the addition of GEXL to the FIBWR code has been verified by comparing the FIBWR calculated CPR's to those reported in the previous Brunswick reload reports. The verifications of FIBWR CPR calculations will be addressed in Section 2.3.4. of this SER.

2.2 CP&L FIBWR Modeling Methods

The modeling aspect of FIBWR code application consists of the core geometric modeling, the determination of coefficients for pressure drop and bypass leakage flow calculations. The reviews are as follows:

2.2.1 Geometric Modeling

Each fuel assembly is modeled in detail including the the inlet orifice, fuel support piece, lower tie plate, heated and unheated rodded regions, grid spacers, water tubes, upper tie plate and exit region. The actual physical dimensions of the fuel, core components and lower internals are taken from the Brunswick plant specific documents and the published GE reports. The CP&L modeling method using one channel to represent one fuel assembly is satisfactory. For input generation for the neutronic calculations, it is necessary to model the core in terms of bundle-by-bundle representations. In other situations, such as the hot-channel analysis, it is sufficient to represent several geometrically similar bundles by one FIBWR channel to reduce the computing cost.

In response to a staff question regarding the core modeling (Ref. 8), CP&L stated its intention to use a "compressed" core representation for most hydraulic applications. In this model, the 75-channel one-eighth core is collapsed into four to six channels depending on the core loading patterns. In general, the fuel channels of the same fuel type in the same region (either the core central region or periphery region) are represented as a average channel, and the fuel channel with the highest relative peaking factor is designated as the hot channel. Sensitivity analysis (Ref. 9) was performed by CP&L which showed that the differences between the 75-channel and the compressed models on hot channel flow, core pressure, and minimum critical power ratio were negligible. Therefore use of the compressed code model for maximum critical power ratio (MPCR) analysis is acceptable when the collapsed model must be sufficiently representative of the core fuel loading pattern and the hot bundle must be represented by a single channel with accurate geometric input data and conservative flow boundary conditions. Since many factors such as radial and axial power profiles and core inlet flow distribution affect the location of the hot bundle, the bundle with the highest peaking factor is not necessarily the hot bundle. Therefore, justification has to be made in the selection of the hot bundle for the MCPR analysis.

Based on the above discussion, use of the compressed core model for MCPR analysis is acceptable.

CP&L also studied the effects of changing the axial node sizes. A 24-node axial power shape was reduced to twelve, eight, and six nodes. The results showed that the hydraulic parameters appeared insensitive to the axial node size with a 0.8% change in the hot channel void fraction and a 0.2% change in the core pressure drop and the minimum critical power ratio. CP&L's intention to use a 24-node scheme for FIBWR applications is acceptable.

2.2.2 Determination of Form-Loss Coefficients

The form-loss coefficients for the orifices, lower and upper tie plates, spacers, and water rod entrance are obtained from General Electric (Ref. 10, 11) and EPRI (Ref. 4) documents. The form-loss coefficients have to be requalified if fuel assembly designs are different from the ones used in the analysis of the three documents cited above. Since the form-loss coefficients given in these references were derived specifically for Brunswick fuel assemblies, their use by CP&L is acceptable.

2.2.3 Determination of Bypass Flow Coefficients

The bypass flow was solved through an empirical correlation in terms of the pressure difference across the leakage path.

$$W = C_1 \; P^{1/2} + C_2 \; P^{C_4} + C_3 \; P^2$$

This approach is acceptable for the steady-state calculations as intended by FIBWR, when the dominating driving force for the bypass flow is the pressure differences across the leakage paths.

The accuracy of using this approach depends on the selection of the coefficants C_1, C_2, C_3, and C_4. The licensee's method of determining the coefficents C_1 through C_4 follows that given in the original FIBWR topical (Ref. 4) with an exception that the flow through each bypass path is expressed as a fraction of the flow through the lower tie-plate holes rather than the flow through the finger spring. The detailed derivations of the coefficients for all bypass paths (there are a total of ten paths) are provided in a response to a staff question. These derivations have been evaluated and found to be adequate for the Brunswick applications. The coefficients for the leakage through the lower tie-plate holes are obtained from a GE document (Ref. 12) and the fraction of each bypass flow is obtained from GE data (Ref.11). Since these data are specifically for the current Brunswick plant design, any future change of fuel design that would affect the bypass paths would require a new analysis for the determination of the coefficients C_1 to C_4. Examples of such changes include the sizes and numbers of the bypass holes in the lower tie plate (LTP) and an LTP with or without finger springs.

Based on the above discussion, determination of the bypass flow coefficients is acceptable.

2.2.4 Hydraulic Models

The hydraulic models used for the Brunswick plant analysis have been evaluated and found satisfactory. The models selected by CP&L include the Blasius single-phase friction factors, homogenous two-phase form-loss multiplier, Baroczy two-phase friction multiplier, and EPRI void model.

The coefficients A and B in the Blasius correlation are obtained from an approved GE topical NEDE-24011-P-A-4-US (Ref. 13) and are therefore acceptable.

2.3 CP&L FIBWR Benchmark

2.3.1 Verification of Pressure Drop Prediction

The FIBWR-calculated pressure drops are compared to the pressure drops obtained from the PI edit of the Brunswick plant process computer. The results match closely. However, instead of the measured data, the PI edit is based on a numerical calculation with given boundary conditions such as core power, total flow, system pressure and inlet enthalpy similar to the FIBWR calculation. Therefore, the PI data are not a truly independent source for benchmark purpose. Since the FIBWR code has been independently verified previously (Ref. 3), the comparison of the FIBWR and PI results is sufficient to verify that the CP&L model setup calculates the pressure drop and other core thermal hydraulic parameters correctly.

2.3.2 Verification of FIBWR Leakage Flow

Correctness of the bypass flow calculation is largely dependent on the selection of the pressure drop coefficients in the bypass flow correlation. As discussed in Section 2.1.3, the CP&L FIBWR modeling on bypass flow is acceptable because the pressure drop coefficients are obtained with an acceptable method. For the purpose of verification, a comparison is made for the FIBWR calculated bypass flow and the bypass flow obtained from the Burnswick plant process computer databook. The PC databook provides cycle specific bypass flow as a function of total flow along the 100 percent rod lines of the power flow map. The result of comparison shows good agreement between the FIBWR calculated bypass flow and those from the PC databook for the cases with power-flow conditions along the 100% rod line. For those cases where the power/flow conditions deviate from the 100 percent rod line, the FIBWR-calculated bypass flow has larger deviations from the estimated bypass flow using the databook bypass/total flow curves.

This is an expected result. For example, a high flow-low power condition would result in a void fraction below that typically expected from operating along the 100 percent rod line. A lower void fraction would result in a higher active flow and a lower bypass flow. Therefore, even though the PC databook curve does not correctly estimate the bypass flow for conditions away from the 100 percent rod line, this data serve to confirm the correct trend in the FIBWR bypass flow prediction.

The comparison is in good agreement for the power/flow condition along the 100 percent rod line, we therefore conclude that the FIBWR bypass flow prediction is correct.

2.3.3 Verification of FIBWR Flow Distribution:

A detailed eighth-symmetric, 75-channel FIBWR model was used in the bundle flow distribution calculations. The results matched those of the plant computer closely. Comparison of the FIBWR results with the process computer results for power ranging from 50% to 99% and flow from 39% to 99% showed that the RMS differences on the bundle flows were less than 1.3%. This verification, together with the independent flow rate comparisons given in the EPRI FIBWR report, indicate the 75-bundle FIBWR model as set up by CP&L was satisfactory in predicting core flow distributions.

2.3.4 Verification of FIBWR CPR Methods

The GEXL (General Electric Report NEDE-25422) critical power correlation was added to FIBWR for use in CP&L's steady-state CPR calculations. The FIBWR hot-channel model for Brunswick Units 1 and 2 were established. The CPR values were calculated for different power levels. For verifications of the FIBWR CPR calculation, more than 30 cases were run and compared to the CPR's obtained from the Brunswick Units 1 and 2 reload analysis (Ref 14, 15, 16, 17, 18, 19, 20, 21) performed previously by GE. These data sources are provided by CP&L in response to a staff question. The results of the comparison, shown on figure 7 of the topical report, show that the CPR's calculated by FIBWR and previous reloads agree very well with a standard deviation of 0.5%

Since the issuance of the topical report, CP&L has revised the CPR calculations slightly due to updated information from General Electric on hot-channel boundary conditions. The revision has a negligible effect on the conclusions made by CP&L on the CPR verifications. With more than 30 data points showing consistent agreement with the General Electric CPR results, it was concluded that the GEXL correlation was correctly installed in the FIBWR code.

2.4 FIBWR Applications

CP&L proposed to use FIBWR in the following applications to the Brunswick plant:

1. Calculation of the bypass flow splits for the system transient analysis code (RETRAN).

2. Hot bundle analysis of slow transients.

3. Hot bundle initial conditions for system transient evaluations.

4. Calculation of steady-state thermal-hydraulic core conditions for use in the nodal simulator (PRESTO-B), training simulator, and plant process computer.

5. Investigation of core anomalies (e.g., local power peaks, flow maldistribution).

6. Calculation of pressure drops across internal components, such as channel walls, core support plate, and core shroud.

7. Bypass boiling analysis.

8. Evaluation of CPR-power relationships.

The FIBWR model setup and the approach used by CP&L are acceptable for the above listed applications except for item 2 where the term slow transient requires clarification.

In response to a staff question, CP&L maintains that the steady state for each case FIBWR code can be used to analyze the type of transients in which the heat deposition rate into the coolant due to conditions from the clad surface to moderator can be conservatively represented by the heat generation rate in the fuel. Therefore, for the hot bundle analysis of slow transients, justification has to be given for using the steady-state FIBWR code for transient applications. As it is used for transients, it should be justified for each specific case. It should be ascertained that the use of the steady-state model for transient simulation is conservative, or that the rate of change of core power is not significant, i.e., the time constant for power decrease is in the order of at least ten times larger than the time constant for the fuel temperature change. Also for hot bundle CPR analysis, necessary uncertainty factors, such as those in power level, inlet temperature, radial and axial peaking factors, and system pressure should be included to yield the required confidence level in the final MCPR calculation. Justification should be given to the uncertainty factors selected.

With the above clarification, item 2 is acceptable.

3.0 SUMMARY

We have reviewed the topical report and the results are summarized below:

1. Compression of the reactor core into a few channels is acceptable for the hot channel MCPR analysis. Care must be taken to assure that the collapsed model is sufficiently representative of the core fuel loading pattern and that the hot bundle is represented by a single channel with accurate geometric input data and conservative flow boundary conditions. Since many factors such as radial and axial power profiles and core inlet flow distribution affect the location of hot bundles, the bundle with the highest peaking factor is not necessarily the hot bundle. The selections of the hot bundle for the MCPR analysis must be justified.

2. As discussed above, the form-loss coefficients for the orifices, lower and upper tie plates, spacer, and water rod entrace must be justified each time when the design of the fuel assemblies is changed

from the current Brunswick fuel design upon which the existing coefficients are based.

A similar procedure applies to the calculation of the bypass flow coefficients (Eq. 1 of topical report), namely, any future change of fuel design that would affect the bypass paths requires a new analysis for the determination of the coefficients C_1 to C_4.

3. The axial noding scheme using 24 nodes (approximately 6 in.) is acceptable for the calculations of the pressure drop, void and enthalpy distributions, and the MCPRs.

4. The intended FIBWR applications listed in Sections 1.2 of the topical are acceptable.

   A. For the hot bundle analysis of slow transients justification has to be given for using the steady-state FIBWR code for transient applications. As it is used for transients, it should be justified for each specific case. It should be ascertained that the use of steady-state model for transient simulations is conservative, and that the rate of change of core power is not significant, i.e. the time constant for power decrease is in the order of at least ten times larger than the time constant for the fuel temperature change.

   B. For hot bundle critical power ratio (CPR) analysis, necessary uncertainty factors, such as those in power level, inlet temperature, radial and axial peaking facors, and system pressure should be included to yield the required confidence level in the final minimum CPR (MCPR) calculation. Justification should be given to the uncertainty factors selected.

4.0 REGULATORY POSITION

We have reviewed the topical report, NF-1583.04, "Verification of CP&L Reference BWR Thermal Hydraulic Models Using the FIBWR Code," and based on the above review, we conclude that it is acceptable for reference for the Brunswick reload thermal hydraulic analysis.

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