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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW OF RELOAD ANALYSIS METHODS

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

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1.0 Introduction

The Vermont Yankee Nuclear Power Corporation (VYNPC or the licensee) submitted for NRC staff review technical reports describing methods to be used for analysis of the physics and thermal-hydraulic performance of the Vermont Yankee core.

These reports are:

1. YAEC-1232, "Methods for the Analysis of BWRs, Lattice Physics" 12/31/80 (Reference 11).
2. YAEC-1238, "Methods for the Analysis of BWRs, Steady State Core Physics", 3/24/81 (Reference 12).
3. YAEC-1239P, "Methods for the Analysis of BWRs, Transient Core Physics", 8/19/81 (Reference 13).
4. YAEC-1299P, "Methods for the Analysis of BWRs, Transient Critical Power Ratio Analysis", 3/30/82 (Reference 14).
5. YAEC-1234, "Methods for the Analysis of BWRs, Steady-State Core Flow Distribution Code (FIBWR), 12/31/80 (Reference 2).

This evaluation describes the staff's review of these reports.

2.0 Discussion

The analytical methods described in the above cited reports have previously been used in analyzing Vermont Yankee core performance during Cycle 9 operation. At the time of startup of the Vermont Yankee plant for Cycle 9 operation, the review of these methods had progressed significantly, but was not yet complete. The review was sufficiently complete, however, to allow the staff to approve the use of these methods to support operation of the Vermont Yankee plant during Cycle 9 operation to a cycle exposure of end-of-cycle (EOC) minus 2000 MWD/T (Reference 15). The Review described in this evaluation addresses the use of these methods to analyze the Vermont Yankee core performance throughout the present fuel cycle (Cycle 9) and during future reload fuel cycles.

3.0 Evaluation

3.1 Lattice and Steady State Physics

The licensee has submitted two topical reports which describe its methods of obtaining certain core physics parameters for use in the design of reloads for the Vermont Yankee Plant. These reports are:

YAEC-1232, "Methods for the Analysis of Boiling Water Reactors - Lattice Physics", dated 12/31/80, and

YAEC-1238, "Methods for the Analysis of Boiling Water Reactors - Steady State Core Physics", dated 3/24/81.

We were aided in the preparation of this evaluation by our consultants, Brookhaven National Laboratories.

The lattice physics methods are used to obtain the physics parameters of the individual bundles in the core. These parameters are used as input to the code which calculates the steady state core characteristics. Bundle (lattice) characteristics are calculated by the CASMO code and the core characteristics with the SIMULATE code.

3.1.1 Lattice Physics Methods (CASMO)

CASMO is a two-dimensional, multigroup transport theory code for the calculation of eigenvalues, spatial reaction rate distributions, and depletion of pin cells for BWR and PWR fuel assemblies. Provision is made for treating the important features of each type of assembly. For BWR assemblies these include: cruciform control rods containing cylindrical absorber elements, water gaps, boron curtains and burnable absorbers in fuel rods.

YAEC-1232 contains a brief description of the CASMO code and references documents which give more complete descriptions. This code is similar to the CPM code which is part of the ARMP code package which is distributed by EPRI and is widely used. Major differences include the input-output routines and the transport calculation. CPM uses a collision probability formulation while CASMO uses a transmission probability formulation which requires much less computation time. The two techniques both use integral transport theory and give essentially identical results in typical applications.

The nuclear data library for CASMO is a 25 or 69 group data set based on the Evaluated Nuclear Data File (ENDF/B-III). Some small corrections to this file have been made to take into account later data. In particular, the well-known anomaly in the U-238 resonance capture has been eliminated by adjusting the parameters to correct the infinite dilution resonance integral.

The calculation of lattice properties proceeds by successive stages of homogenization. First pin cells, consisting of a single fuel pin surrounded by clad, moderator and a fourth region representing the surrounding assembly are calculated and effective pin cell cross sections are obtained. These calculations are performed in 69 or 25 groups. Calculations are performed for each pin type in the assembly. For BWR calculations the next step consists of modeling the complete assembly as a cylinder consisting of rings of the different pin types and calculating the effect of pin location on its spectrum. These calculations are performed in 25 groups. The final step consists of constructing an x-y model of the assembly where each pin cell is represented explicitly. The pin cell cross-sections are collapsed to the 7 to 12 groups used in the x-y calculation making use of spectral correction factors obtained from the cylindrical calculation. The whole assembly is then represented by a set of two-group cross-sections which are used as input to the SIMULATE code.

Depletion calculations are performed in the 7 to 12-group x-y model to obtain the heavy nuclide and fission product concentrations as a function of burnup.

The suitability of the CASMO code for the Vermont Yankee core analyses has been verified by comparison with various experiments and certain aspects of the calculations have been verified by comparison with higher order (Monte-Carlo) calculations. Comparison with uniform pin cell criticals of eigenvalues calculated by CASMO show that this quantity is calculated to within one percent at the one-sigma level. No trends were noted over the range of enrichment, lattice pitch and boron concentration values of the experiments which encompassed values used in Vermont Yankee. Depletion (burnup) calculations performed by CASMO were compared to experimental results obtained from the Yankee Core Evaluation Program.

In particular the isotopic ratios of the various uranium and plutonium isotopes were compared for fuel rods having burnups in the range from approximately 1.2 to 31 GWD/MTU. The comparisons showed that CASMO predicts isotopic concentrations to within the experimental uncertainty.

In order to test the treatment of strong adsorbers, comparisons were made to a series of experiments which contained poison slabs between assembly mockups. Cross sections prepared by CASMO were used in a fine mesh diffusion theory calculation to obtain the k_{eff} (criticality) value for the critical configuration. The results showed that k_{eff} was calculated to within 1.6 percent.

In order to test the treatment by CASMO of gadolinia bearing fuel rods and control rods, comparisons were made to KENO-IV (Monte-Carlo) calculations. Care was taken to assure consistency between the two calculations -

identical geometries, cross-section libraries, temperatures and nuclide densities were used. Comparisons were made of infinite multiplication factors and power distributions. The agreement for both quantities was within two standard deviation of the statistical uncertainty in the KENO calculations. The YAEC-1232 report also presents as background information data of a similar nature that were used in Sweden to qualify the code. Similar results were obtained.

3.1.2 Steady State Core Physics Methods (SIMULATE)

The SIMULATE code, described in YAEC-1238, is a three-dimensional, steady state, nodal code for the calculation of PWR and BWR core characteristics. Power distributions, cycle burnup, control rod patterns and xenon transients may be calculated. For BWR applications the core is represented by 24 or 25 axial nodes each of which is one assembly pitch in lateral dimensions. (This means that each node is approximately a six-inch cube). Nodal cross sections are prepared by the CASMO code described above to be used as input to SIMULATE. CASMO calculations are performed for several values of the various independent variables on which cross-sections depend (e.g. fuel type, burnup, assembly void content, etc.) and are entered as tables or analytical fits. Fits or tables are entered to permit two-group cross sections to be obtained. Nodal properties for use in the SIMULATE calculation are computed from these cross-sections.

YAEC-1238 includes a description of the procedures used for cross-section fitting. These are similar to those usually used in nodal codes. The one group constants used in the transport calculations are obtained by standard diffusion theory techniques.

The internodal neutron transport calculation is performed with the coarse mesh diffusion theory formalism developed by Borresen. Transport into and out of a particular node is assumed to take place to and from the six neighboring nodes only. The nodal faces on the core boundary are treated by the use of albedos. Interfaces between nodes of different compositions (e.g. controlled and uncontrolled nodes) are treated by use of a thermal leakage correction.

The procedures for solving the neutron balance equation are described. A multilevel iteration scheme is employed. Inner and outer source iterations are performed to obtain a converged source (flux) distribution. This distribution is then used to alter the core flow distribution (which depends on the power distribution), a new source is calculated and the process is repeated until both source and void iterations are converged. This procedure is similar to that used in other nodal calculation codes.

The algorithm and techniques employed for the calculation of power and time dependent xenon and samarium concentrations are presented. They are standard for these calculations. The various core depletion options are also described. These include both depletion with timestep average power distributions and the Haling end-of-cycle power-exposure iteration.

Verification of the suitability of the SIMULATE code for use in the analysis of Vermont Yankee cores is provided. Calculations by the CASMO-SIMULATE code combination are compared to measured core parameters

for the first seven cycles of Vermont Yankee and to some gamma scan measurements performed on the Quad Cities reactor. Further verification of some details of the calculations is provided by comparison of SIMULATE results with those from a fine mesh diffusion theory (PDQ) calculation with discrete modeling.

Calculations of the critical eigenvalue (K_{eff} value) for both hot and cold states were made for various exposures during the seven Vermont cycles. These calculations showed that the standard deviations of the results were 0.0027 and 0.003 for the hot and cold eigenvalues respectively. The cold eigenvalue had a bias of -0.011 and the hot bias was +0.0019.

Comparisons were made of calculated and measured traversing incore probe (TIP) traces at several exposures during each of the first seven cycles at Vermont Yankee. In addition, comparisons were made between end-of-cycle assembly exposure distributions calculated by SIMULATE and those obtained from the process computer. Evaluation of this body of data leads to the conclusion that peak-to-core average core TIP trace ratios may be calculated to within a standard deviation of 6.4 percent.

In order to investigate the treatment of neutron transport between nodes of very different properties a comparison was made between SIMULATE and and fine-mesh diffusion theory (PDQ-7) calculations. Nine assembly (3X3) arrays were calculated in which the central assembly had very different parameters from the surrounding eight. Differences in exposure and void content were investigated. Quarter core calculations were performed to investigate the effect of controlled and uncontrolled nodes.

The power distributions were the same to within two percent and the eigenvalues to within 0.1 percent. The ability of SIMULATE to calcualte transient xenon effects was verified by performing core follow calculations for the early part of Cycle 3 of Vermont Yankee. These calculations showed that eigenvalue and power distributions were predicted to within essentially the same accuracy as those at steady state conditions, except for low power conditions coupled with major control rod pattern changes.

The manner in which the SIMULATE code is used to obtain core wide reactivity coefficients is discussed in Section 3.2 of this report.

3.1.3 Summary of Evaluation

The following discussion summarizes our evaluation of the CASMO and SIMULATE codes.

The CASMO code is described sufficiently to permit the conclusion that the code is state-of-the-art. References to more complete descriptions are given. The data base used for the cross-section library is the industry standard Evaluated Nuclear Data File (ENDF/B in Version III). Corrections have been made to the data base to correct for known deficiencies in this version. Use of this data base is standard industry practice. The calculation procedure employed in the use of the code is described. This procedure is state-of-the-art and is acceptable.

Verification of the ability of CASMO to perform lattice calculations is provided by comparing CASMO results to experiments and to higher order (Monte Carlo) calculations. Measured data from critical experiments,

both uniform pin cell lattices and lattices containing lumped poisons - and from experimental studies of isotopic concentrations as a function of burnup were used. This data base is similar to that routinely used for such purpose. Comparison to Monte Carlo calculations were made for the purpose of verifying some details of the calculations in the vicinity of gadolinia bearing rods. The description of CASMO including the data base used and verification procedures are acceptable.

Based on the description of the algorithms and calculation procedures employed in the SIMULATE code, we conclude that these are state-of-the-art. These include the modeling of the core, the production of the cross-section tables and curve fits, the nodal neutron transport algorithm, the iteration technique used to solve the neutron balance equation, the algorithms used for transient xenon and samarium, and the techniques used for core depletion.

Verification of the SIMULATE code was done by comparing calculated results to measured core parameters for the first seven cycles of the Vermont Yankee reactor and to gamma scan measurements performed in the Quad Cities reactor. Some calculation details were verified by comparison of SIMULATE results with fine mesh diffusion theory calculations. Based on the data provided, we find that adequate verification has been performed.

Comparisons of calculated and measured critical eigenvalues in hot and cold states show that SIMULATE uncertainties are similar to those obtained by other such calculations. Comparisons between SIMULATE calculations and TIP traces from Vermont Yankee showed that the peak linear heat generation rate is calculated to acceptable accuracy. Based on these comparisons, we conclude that the CASMO-SIMULATE code package is acceptable for use in calculating input constants for the process computer.

The review of topical reports YAEC-1232 and YAEC-1238 has been conducted within the guidelines provided for analytical methods in the Standard Review Plan, Section 4.3.

3.1.4 Conclusion

Based on our review of topical reports YAEC-1232 and YAEC-1238 we conclude that they are acceptable for reference in licensing actions for the Vermont Yankee reactor. Such reference may be made for purposes of describing the lattice physics and steady state physics methods employed in analyses for Vermont Yankee and for analysis of xenon transients. These reports may also be used as references for uncertainty values in the following parameters.

- Cold critical eigenvalues and shutdown margins
- Hot critical eigenvalues
- Gross core peaking factor
- Peak linear heat generation rates
- Burnup distributions
- Transient xenon effects.

3.2 Transient Core Physics

3.2.1 Description of YAEC-1239P

As part of a series of reports documenting the Yankee physics methodology, the Yankee Atomic Electric Company has submitted a proprietary topical report YAEC-1239P, entitled, "Methods for the Analysis of Boiling Water Reactors - Transient Core Physics." This report describes the methods and procedures employed for computing core physics input parameters for the RETRAN transient systems analysis code.

In view of the proprietary nature of the report, it is described below in general outline only. The RETRAN code uses point kinetics to describe the transient behavior of the core. The report discusses the neutronic input required for RETRAN and then describes the procedures used to provide input which minimizes the error resulting from the use of point kinetics. These procedures include the use of the SIMULATE nodal analysis code to obtain three-dimensional neutronic parameters for base and perturbed cases, the reduction of these parameters to one-dimensional ones, and finally, the reduction of the one-dimensional parameters to reactivities for use in RETRAN point kinetics calculations. Care is taken to preserve the eigenvalue, the two-group fluxes and the average axial power distribution in the reduction to one-dimension and the eigenvalue in the reduction to point kinetics. Detailed mathematical developments of the techniques used in the reductions are presented in appendices to the report.

Procedures are also described for accommodating the differences in the void model used in SIMULATE and RETRAN. This is accomplished by introducing a second term in the correlations with void content.

The qualification and verification of the transient core physics method are based on adherence to "first principles" in defining the procedures and performing the analyses, comparison of calculated results with experiments, and sensitivity studies. In addition, the cross-section preparation code, CASMO, and the three-dimensional nodal code, SIMULATE, have been verified against experiments to assure that the basic inputs are appropriate.

Comparisons of measured and calculated core response are presented for the Peach Bottom reactor turbine trip tests. The core power response was converted to a core reactivity response as a function of time during the transients and compared to the calculated response. The various components of the calculated reactivity are also presented. These comparisons show that the time behavior of the response is accurately calculated and the total reactivity insertion is slightly over-predicted.

Investigations of the Doppler and void weighting factors are presented. The behavior of these factors as a function of core conditions is then compared to what would be expected from physical considerations. The scram reactivity function is calculated by both static and dynamic methods and it is shown that the static method is conservative for the turbine trip transients. In order to check the validity of using single values of the kinetics parameters (beta effective and prompt neutron generation time), during the transient a series of calculations of these quantities as a function of core conditions was performed. These calculations showed that the kinetics parameters are insensitive to core conditions and that the changes would have negligible effect on the core transient behavior.

3.2.2 Evaluation

The following discussion summarizes our evaluation of the subject topical report.

The basic input for the core transient analysis is obtained from CASMO-SIMULATE calculations. These codes have been previously reviewed and approved by the staff and are acceptable for this purpose. The three-dimensional core description which results from these calculations - eigenvalue, cross-sections, power distributions, etc. - is then reduced to the point kinetics form used in the Yankee RETRAN calculation. We have reviewed the algorithms and procedures used in this reduction and conclude that they are state-of-the-art.

The comparison of the calculated core reactivity response with that measured in the Peach Bottom turbine trip tests provides acceptable verification of the suitability of the techniques, procedures, and algorithms used to calculate core transient behavior. The sensitivity studies also support this conclusion.

The comparison of static and dynamic scram curves supports the conclusion that the use of the static curve generated at pretransient conditions is conservative. The results of the study of kinetic parameters as a function of core conditions support the use of constant values for these parameters.

The review of topical report YAEC-1239P has been conducted within the guidelines provided for analytical methods in the Standard Review Plan, Section 4.3.

3.2.3 Conclusion

Based on our review of topical report YAEC-1239P, which is decribed above, we conclude that it is acceptable for reference in licensing actions for the Vermont Yankee reactor. Such reference may be made for the purpose of describing the core model used in the RETRAN code and the algorithms and procedures used to obtain the core physics parameters used with the model. These parameters include reactivity coefficients, scram curve, and the kinetics parameters. This review has not considered the procedures used to define margins for licensing actions.

3.3 RETRAN

By letter dated March 31, 1982, Vermont Yankee Nuclear Power Corporation submitted YAEC-1299P, "Methods for the Analysis of Boiling Water Reactors Transient Critical Power Ratio Analysis (RETRAN-TCPYA01)," for staff review. This methodology will be used for the transient critical power ratio (CPR) calculation for the Vermont Yankee plant reactor core reloads. The methodology consists of two steps: (1) RETRAN is used to perform hot channel thermal hydraulic calculations to determine the time-dependent axial thermal-hydraulic conditions such as enthalpy and flow rate and (2) TCPYA01 is used to read the RETRAN output of nodal thermal hydraulic conditions and bundle pressure and corresponding saturated liquid and vapor enthalpies to perform minimum CPR calculations. Therefore, this methodology review consists of two parts: (a) the acceptability of the RETRAN code to perform BWR hot channel thermal hydraulic calculations and (b) the acceptability of TCPYA01 with incorporation of the GEXL correlation to perform CPR calculations.

3.3.1 EVALUATION

RETRAN Hot Channel Analysis Review

We reviewed the RETRAN hot channel analysis method with technical assistance from Argonne National Laboratory (ANL). We and ANL have concluded that the nodalization used in the RETRAN analysis is appropriate; the selection of options (such as friction multiplier, void fraction model and automatic time step control, etc.) and input ot RETRAN provide a resonable and adequate representation of the thermal hydraulics and the results of these calculations can be used to determine an acceptable set of input and initial conditions for the critical power ratio calculation. Therefore, the use of the RETRAN computer code to perform the hot channel thermal hydraulic calculations is acceptable.

TCPYA01 CPR Calculation Review

TCPYA01 utilizes the GEXL critical quality-boiling length correlation to calculate CPR at each time step during a transient with inputs provided by the RETRAN hot channel transient analysis results. The method for the calculation of minimum CPR is an iterative process. At each time step, RETRAN generates axial nodal enthalpy and mass flow rate distributions, and the bundle pressure and corresponding saturated liquid and vapor enthalpies. These conditions are input to the TCPYA01 code. For each axial node, TCPYA01 estimates a CPR which, in turn, is used to calculate the nodal quality distribution and axial location of the boiling boundary at the estimated critical power level. The boiling length is then calculated and the critical quality of the node is calculated using the GEXL correlation based on the nodal mass flux, bundle pressure and the boiling length. From the nodal quality and the critical quality comparison, the value of CPR from the previous iteration is revised. The revised CPR is again used to determine the nodal quality, critical quality and a further revised CPR. This iterative process is continued until the CPR interations converge and the CPR of the node is thus determined. This process is performed for all axial nodes and the hot channel minimum CPR is defined as the lowest nodal CPR calculated for the time step.

The staff has reviewed this CPR calculation procedure and concluded that the method is correct and acceptable.

CPR Methodology Qualification Review

To qualify the overall RETRAN-TCPYA01 CPR methodology, the licensee has performed benchmark analysis against both steady state and transient boiling transition (BT) data from the ATLAS 16-rod electrically heated test sections. A total o[illegible] steady state data points were analyzed with RETRAN-TCPYA01. The result[illegible] that all but one of the steady state critical power data are predic[illegible]hin ±5% (the other point is predicted within 6%). The resulting me[illegible] CPR is 1.006 and the standard deviation is 1.86%. This result dem[illegible]he adequacy of the GEXL MCPR limit of 1.07.

A total of 27 transient tests were analyzed using RETRAN-TCPYA01. These transients consist of 13 flow-decay-at-constant-power cases, 12 flow-and-power-decay cases and two flow-and-power-increase cases. For the 13 flow-decay at constant power transients, the predicted times to BT (CPR=1.0) are generally shorter than the experimental times to BT except for two cases. In one of these two cases, the predicted time to BT is slightly longer than the experimental time to BT. In the other case, the boiling transition is not predicted but the MCPR is predicted to be 1.019 which is well within the MCPR limit of 1.07 for the GEXL correlation. For the two flow-and-power-increase transient cases, the predicted times to BT are also shorter than the experimental times to BT. For the 12 flow-and-power decay transient cases, the RETRAN-TCPYA01 analyses either predict longer times to BT or do not predict BT. For those cases where BT is not predicted, the predicted MCPR are all within the GEXL correlation MCPR limit of 1.07 except for one case having a MCPR of 1.081. Several factors may have contributed for the less conservative predictions of the flow-and-power decay cases:

(a) For the RETRAN-TCPYA01 transient analyses, the initialization is performed by varying the initial power until the predicted exit quality matches the experimental exit quality. As a result, the initial powers used in the analysis do not match the measured power inputs to the test section. The licensee has indicated that the ratios of the predicted initial power to the measured power for the transient cases are approximately 0.98. This 2% deficit of the predicted initial power level results in a higher CPR prediction.

(b) For the flow-and-power decay transients, the CPR versus time curves are generally more flat. Therefore, the predicted time to BT is very sensitive to small changes in the predicted CPR, and an error of only a few percent in terms of critical power can lead to a large discrepancy between predictions and experiments in terms of time to BT.

(c) There are uncertainties in both the GEXL correlation and experimental data. As indicated by the licensee, uncertainties of up to ±3% in the test section were observed for the flow-and-power decay transients.

Thus, although the RETRAN-TCPYA01 predictions for these transients are less conservative the errors are within the uncertainty of the GEXL correlation with a 1.07 MCPR limit.

Sensitivity Studies

The licensee has performed sensitivity studies of the MCPR predictions with respect to axial nodalization, void-prediction models and time-step size. The results show that a total of 12 nodes to represent the active core region are sufficient. An increase in the number of axial nodes does not significantly change the predicted results. It was also found that the MCPR prediction is relatively insensitive to the void model used. Therefore, either the homogeneous equilibrium model or the EPRI void model can be used to calculate transient MCPR. A study of the time step sensitivity indicated that the predicted results are relatively insensitive to time steps in the range of time step size investigated (0.001 seconds to 0.05 seconds). This range of time step sizes is similar to those used in the proposed licensing transient simulation.

RETRAN-TCPYA01 Application

The RETRAN hot channel modeling for Vermont Yankee CPR calculations consists of fourteen (14) control volumes (axial nodes), thirteen (13) flow junctions and twelve (12) heat conductors. The active core region is represented by 12 nodes. Since the sensitivity studies performed by Vermont Yankee have determined that an increase in the number of axial nodes does not significantly change the predicted CPR results, this modeling is acceptable.

For the transient hot channel CPR calculations, the time-dependent boundary conditions such as pressure, inlet enthalpy and normalized power for the hot channel are obtained from the system level RETRAN transient analysis. The initial flow to the hot channel is obtained from the FIBWR calculation by imposing the initial core pressure drop obtained from the system level calculation. The result of the RETRAN hot channel calculation of nodal flow and enthalpy are input to TCPYA01 for the calculation of CPR. Since the GEXL correlation is not directly incorporated into the RETRAN code, it is not

possible for RETRAN to predict boiling transition using the GEXL correlation. Therefore, the RETRAN-TCPYA01 CPR calculation methodology is valid only up to the time at which boiling transition is predicted by the TCPYA01/ GEXL correlation.

RETRAN has a fairly simple fuel pin model where the pellet-cladding gap conductance is input as a constant value. In reload applications, a conservatively large gap conductance should be used to predict lower CPR to account for the gap conductance uncertainty. The input gap conductance should be obtained from the analysis of the most severe transient which will provide a bounding gap conductance for calculation of other transients. In addition, the uncertainty of other input parameters should be considered and conservative values that result in lower CPR should be used for reload analyses.

CONCLUSION

We have reviewed the RETRAN- TCPYA01 CPR methodology. The RETRAN hot channel modeling is found acceptable to perform transient hot channel thermal hydraulic calculations and the TCPYA01 code is acceptable for CPR calculations. However, in the use of RETRAN, conservative values that account for uncertainties of input parameters should be used for hot channel analyses.

3.4 FIBWR

3.4.1 Description of FIBWR

By letter dated December 31, 1980 (Ref. 1), the licensee submitted the FIBWR topical report (Ref. 2) for the staff review to support the Vermont Yankee reload analysis methodology for Cycle 9 operation.

FIBWR is a steady-state thermal hydraulic analysis code which determines the flow and void distributions for a given power distribution and inlet flow condition in a BWR core. The total flow entering the lower plenum splits into active components flowing through the fuel channels, and bypass or leakage components flowing through the interstitial regions surrounding the fuel channels. The flow distribution in the core is determined through the balance of pressure drop across the core and the total inlet flow rate. The pressure drop is calculated by integration of the momentum equation. The total pressure gradient is comprised of the spatial acceleration, friction, gravitational and local loss components. Void distribution affects the pressure drop due to its effects on the flow density and on the two-phase multipliers on friction and local shock losses. The energy equation is solved to determine the thermodynamic equilibrium quality, which, in turn, is used to calculate flow quality and void fraction along each channel.

3.4.2 Evaluation

Pressure Drop Calculations

The review of the FIBWR code included the method of solution, the constitutive relationships, data comparison and overall code benchmark evaluation.

A flow channel is divided into many axial nodes. The pressure drop in each axial node is calculated by integration of the momentum equation. The total pressure gradient is comprised of the spatial acceleration,

gravitational, friction and local loss components. The staff has reviewed the equation of each pressure drop component and concludes that they are based on widely-accepted standard techniques.

The staff did question (Ref. 3) the equation for acceleration pressure drop due to flow area change in the single phase unheated region, equation 2-3, which fails to identify the location where the mass velocity, G, is calculated. The licensee has responded (Ref. 4) that G is actually G_2 (i.e., the mass velocity evaluated at the downstream flow area) and the staff finds it correct. In the two-phase flow region, the momentum density is used in the acceleration pressure drop calculation. The momentum density, which is a pseudo-density, was originally developed by Meyer (Ref. 5) and has been widely used in the industry. Therefore, the acceleration pressure drop calculation is acceptable.

The single-phase friction factor is calculated by the Blausius expression where a smooth surface is assumed. The licensee indicates (Ref. 4) that the surface roughnesses of the fuel cladding and other flow paths are on the order of 30 micro inches. In the normal operating range, where the Reynolds number is on the order of 2×10^5, the friction factor predicted by the Blausius equation is within 2% of the Moody friction factor and is, therefore, acceptable. The two-phase friction multiplier is calculated by the modified Baroczy correlation used in the RETRAN code. This correlation has been compared to experimental data and represents the current state of the art.

For the two-phase local shock loss, FIBWR uses a modified homogeneous multiplier with an empirical constant β as an input parameter. However, in the Vermont Yankee design analysis the value of β is set to 1.0, which essentially represents the homogeneous multiplier. As stated by Lahey and Moody (Ref. 6), extensive experimental data taken on a wide range of grid-type spacers have indicated that the homogeneous multiplier does a fairly good job of correlating the two-phase pressure drop data.

The use of the homogeneous multiplier in the design analysis associated with an adjusted form loss coefficient in the upper tie plate to match the measured pressure drop at Vermont Yankee is a reasonable approach for steady-state flow distribution calculations. We find the FIBWR pressure drop calculation methods acceptable.

Void Fraction Calculation

FIBWR uses a simplified EPRI void model for void fraction calculations. The original EPRI subcooled void model (Ref. 7) is a mechanistic model for flow boiling in vertical geometries and is qualified against steady-state void formation data. The staff did not review this model. The simplified EPRI model is not explicitly presented in the FIBWR topical, but is described in the EPRI-FIBWR report (Ref. 8). The model uses a quadradic equation to determine the coolant temperature, $T_{departure}$, associated with subcooled void formation. This equation was derived from the assumptions that vapor generation is a result of wall heating, and that the total wall heat flux at any point in the heated channel is comprised of forced convection and nucleate boiling calculated by Thom's correlation. The onset of subcooled boiling is determined by setting the vapor generation rate equal to zero. After the onset of void detachment, the flow quality is related to the thermodynamic quality by a hyperbolic tangent profile fit. The void fraction is then determined by a drift flux model. The distribution parameter, Co, and drift velocity, V_{gj}, are dependent upon the flow regime and, therefore, dependent upon pressure and void fraction. A modified Bancoff-Jones correlation is used for the distribution parameter. The drift velocity, V_{gj}, is given by a modified bubble terminal rise velocity where the constant k_3 is replaced by a void-dependent expression. Since both Co and V_{gj} are void-dependent, the void fraction calculation is done by an iterative technique.

Since there is no data to benchmark against each individual constitutive equation, the simplified EPRI void model is verified on the overall

basis. The licensee has made a comparison of the void prediction of the simplified EPRI model with the homogeneous Dix and Zuber-Findlay void models. The result (Figure 3.3, Ref. 2) shows that the simplified EPRI model lies between these models. Furthermore, the model is also compared with the FRIGG Loop (Ref. 9) data and shows good prediction results (Figures 3-6, 3-7, 3-8, 3-9 of Ref. 2). Therefore, the staff concludes that the modified EPRI model is acceptable for use in FIBWR.

Solution Technique

The flow distribution calculation is done by an iterative technique. On the inner iteration, core pressure drop is estimated, and the flow in each channel is iterated until the desired pressure drop is converged. Then the flows from each of the channels and bypass regions are summed up to obtain the total core flow, which is compared to the required core flow. If the total core flow is not convergent, the outer iteration is done by readjusting the core pressure drop and repeating the inner iteration. This iterative process is a routine pactice and is acceptable.

Code Verification

In order to verify the numerical procedure coding of the FIBWR code, the licensee has performed a comparison of the FIBWR results with the analytical solution of a simple problem. The case is a homogeneous equilibrium two-phase flow in a vertical heated tube with uniform axial heat flux. Using the optional models in FIBWR to match the model used in the analytical solution, the results show excellent agreement.

The licensee further performed a comparison between FIBWR and COBRA-III C (Ref. 10) by turning off the transverse flow in COBRA and selecting comparable models in the two codes. The results show agreement to within 10%

as shown in Figures 4-3, 4-4 and 4-5 of Reference 2. In response to the staff questions on the cause of the 10% discrepancy between the two codes, the licensee has rerun the COBRA-III C using coolant properties obtained from FIBWR and increasing the number of axial nodes in COBRA input to match the FIBWR runs. The results show very good agreement between the two codes as shown in Table 1 and Figure 4 of Reference 4. Furthermore, the licensee has performed a benchmark comparison of the FIBWR code aginst Vermont Yankee plant data. The results show good agreement as shown in Table 5-8 of Reference 2.

Conclusion

Based on the observations described in the above sections, we conclude YAEC-1234 is acceptable for referencing in licensing actions for the Vermont Yankee plant.

4.0 Summary and Conclusions

The staff has completed its review of the following reports describing methods to be used for analyses of physics and thermal-hydraulic performance of the Vermont Yankee core: YAEC-1233, YAEC-1238, YAEC-1239P, YAEC-1299P, and YAEC-1234. We conclude that the use of the methods described in these reports in analyzing the Vermont Yankee core performance is acceptable for reference in licensing actions. In the use of RETRAN-TCPYA01 (YAEC-1299P), conservative values that account for uncertainties of input parameters should be used for hot channel analysis.

REFERENCES

1. Letter from R. L. Smith (Vermont Yankee Nuclear Power Corporation) to USNRC, "Submittal of Documentation for Core Relad Licensing Methods," dated December 31, 1980. Docket No. 50-271.
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3. Letter from T. A. Ippolito (USNRC) to R. L. Smith (VYNPC) dated November 10, 1981.
4. Letter from R. L. Smith to USNRC, "Response to NRC Questions Regarding FIBWR Computer Code," dated November 6, 1981. Docket No. 50-271.
5. J. E. Meyer, "Conservation Laws in One-Dimensional Hydrodynamics," Bettis Technical Review, WAPD-BJ-20, Westinghouse, 1960.
6. R. T. Lahey, Jr. and F. J. Moody, "Thermal-Hydraulics of a Boiling Water Nuclear Reachor," American Nuclear Society.
7. G. S. Lellouche, B. A. Zolotar, "A Mechanistic Model for Predicting Two-Phase Void Fraction for Water in Verical Tubes, Channels, and Rod Bundles," through letter from R. L. Smith (VYNPC) to USNRC, "Vermont Yankee Reload Licensing Method", dated September 3, 1981. Docket No. 50-271.
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9. O. Nyland, K. M. Becker, et al., "Hydrodynamics and Heat Transfer Measurements on a Full Scale Simulated 36-Rod Marvikken Fuel Element," ASEA-ATOM FRIGG LOOP, Sweded, R4-494/RL-1154.
10. D. S. Rowe, "COBRA-III C: A Computer Program for Steady-State and Transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements," BNL, March 1973.

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13. YAEC-1239P, "Methods for the Analyses of BWRs - Transient Core Physics, August 1981, Yankee Atomic Electric Co.

14. YAEC-1299P, "Methods for the Analyses of BWRs - Transient Critical Power Ratio Analysis, March 1982, Yankee Atomic Electric Co.

15. License Amendment No. 70 to Facility License No. DPR-28, dated November 27, 1981.