UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO GPU NUCLEAR COPRORATION TOPICAL REPORT
TR-021, REVISION 0
"METHODS FOR THE ANALYSIS OF BOILING WATER REACTORS STEADY STATE PHYSICS"
GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION
DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated March 25, 1986 (Ref. 1), the GPU Nuclear Corporation (GPUN) submitted for review TR-021, Revision 0, "Methods for the Analysis of Boiling Water Reactors Steady State Physics." The information in this report was supplemented by information submitted with References 8 and 9 in response to requests for additional information from the NRC staff and consultants. The review by the staff of this report and supplemental information was performed with the assistance of consultants from Brookhaven National Laboratory (BNL).

As indicated in Reference 1, it is the intent of GPUN to conduct in-house analyses for core related changes to the Oyster Creek Nuclear Generating Station (Oyster Creek) Technical Specifications during Cycle 11, and perform reload core safety analyses for Cycle 12. This report is the second of four submitted by GPUN. A report on lattice physics (TR-020) has been reviewed and approved and two reports related to the analysis of transients (TR-033 and 040) are being reviewed. This report (TR-021) describes the three-dimensional BWR steady state coupled neutronic/thermal-hydraulic modeling using the (neutronic) EPRI-NODE-B and (thermal-hydraulic) EPRI-THERM-B codes, and is referred to as the NODE-B code. The report also provides verification of the accuracy of the calculations with NODE-B by comparisons with measured data.

Both EPRI-NODE-B and EPRI-THERM-B are part of the Advanced Recycle Methodology Program (ARMP) code system (Ref. 4). The NODE-B three-dimensional core simulator code has been developed with the EPRI Power Shape Monitoring System (PSMS) (Ref. 5), an on-line hybrid system which monitors core performance and power distribution. The integrated NODE-B/THERM-B code system of the on-line PSMS has been converted for use by GPUN on the IBM computer for off-line analysis. The modeling and verification of the NODE-B/THERM-B integrated system, used off-line, is the subject of the present technical evaluation.

2.0 DESCRIPTION OF THE METHODOLOGY

The two ARMP codes, EPRI-NODE-B and EPRI-THERM-B, have been integrated into a single code, NODE-B, for the PSMS application. The integrated code is a coupled three-dimensional neutronic and thermal hydraulic model in which a complete calculation consists of a converged set of iterations between neutron source and moderator voids.

2.1 EPRI-NODE-B

The EPRI-NODE-B code is a successor to the FLARE code (Ref. 6). A modified one-group theory model is used in this code. The key input parameters in EPRI-NODE-B are the neutron multiplication, k-infinity, and the migration area, M^2. These parameters are derived from detailed energy and space-dependent calculations for each fuel assembly type and are entered in the nodal calculation as a function of coolant voids and exposure, including the effects of control, coolant temperature, Doppler and xenon. The fuel assemblies are coupled together in EPRI-NODE-B using a transport kernel which is a function of the migration area and the nodal mesh spacing. The transport kernel plays an important role in the nodal calculations since it, along with the local multiplication and leakage factors, is used by the code in the calculation of the three-dimensional power distribution. The code calculates the transport kernel in each node in the horizontal and vertical directions using input constants which are selected such that the results of the basic model calculations are normalized to a more accurate calculation such as PDQ or to measured data.

2.2 EPRI-THERM-B

This code calculates the thermal hydraulic parameters of the core including flow distribution, subcooling, void and quality distributions based on total core power, recirculation flow, power distribution, and feedwater flow and temperature. Since the coolant flow distribution through the core is influenced by the void content and the power level, an iterative calculation is required to determine the power and flow distribution.

The flow distribution is obtained by equalizing the pressure drop across each channel. This calculation starts with an initial guess for the coolant velocity in each channel and the pumphead requirements, and proceeds iteratively until the coolant velocity converges within a specified tolerance. The process is repeated for each channel. When a distribution is obtained for all of the channels, all individual channel flows are summed and compared to the total core flow. The calculation is complete when the summed flow is within a specified tolerance of the total core flow.

The subcooling in the EPRI-NODE-B code is calculated by performing a heat balance in the downcomer and lower plenum regions of the vessel. The single-phase loss coefficients are input to EPRI-NODE-B. These coefficients are corrected during the calculation for the local quality and void condition within the channel. The relative moderator density, a key variable in the representation of the nuclear properties of the core, is determined by calculating the nodal quality from the power and channel flow rates. The Zolotar-Lellouche (Ref. 7) void-quality model is employed in the thermal hydraulic code.

3.0 EVALUATION

The evaluation of this report is based on the review of the methods underlying the NODE-B code and the verification of those methods with measured data. The material for this review includes both the report (TR-021) and the responses to questions (Ref. 8 and 9).

3.1 Modeling

Neutronic Modeling

Material constants including the neutron multiplication and migration area are derived from multigroup fuel assembly calculations and introduced into NODE-B as functions of exposure and moderator voids for both uncontrolled and controlled assemblies. The calculation of the Doppler reactivity effect in each node is based on a square root of fuel temperature dependence with appropriate power and moderator density corrections. These representations are known to be adequate for the analyses intended to be carried out with NODE-B.

Nodal exposure effects are determined by calculating fractional reactivity changes from exposure and void dependent reactivities input in the code. Nodal exposure is updated with each exposure step using the nodal power at the start of each time step. Exposure-weighted voids in each node are computed at the end of each time step. Control history effects on the exposure-weighted voids can be included in the calculation. This exposure reactivity modeling in NODE-B is sufficient for core follow and reload analyses and therefore it is acceptable.

The xenon microscopic cross section is input in NODE-B as a linear function of the moderator density and, together with the nodal thermal flux and power, is used in the evaluation of the xenon number density. The xenon reactivity effect is then calculated in each node. Since the important xenon and iodine effects on nodal reactivity and power are adequately represented in NODE-B, this treatment is acceptable.

Thermal Hydraulic Modeling

The integration of EPRI-THERM-B with EPRI-NODE-B into a single code, NODE-B, eliminates the possibility of errors in transferring data between the two code modules during the neutronic/thermal-hydraulic iteration process.

Starting with an initial guess for the coolant velocity in each channel and an initial guess for the pumphead requirement, the solution to the hydraulic equations is obtained iteratively. The coolant velocity in each fuel channel is varied until it yields a pressure drop which corresponds to the pumphead requirement within a specified tolerance. The individual channel flows obtained from the converged coolant velocities are summed and the resulting flow is compared to the total core flow. If the two flows lie within a specified range the problem is converged. Otherwise, the pumphead is adjusted to reduce the difference between the calculated and specified flow in the input and the entire iterative procedure is repeated.

The fuel assembly pressure drop is obtained as a function of the square of the liquid coolant velocity, the boiling and non-boiling lengths, the friction factor and void conditions. The moderator density is determined from the nodal quality, which is derived from the power and channel flow rates and the steam volume fraction. It is this moderator density which is used in the neutronic/thermal hydraulic iteration to establish the values of the nuclear constants in EPRI-NODE-B.

The methods employed in the thermal-hydraulic calculations are acceptable for representing the steady state behavior of the Oyster Creek core.

Input Model

The input model in NODE-B consists of neutronic and thermal hydraulic data. Basic core and fuel design data, power level, control rod position, nuclear constants, core flow and thermal-hydraulic characteristics are specified in the input. Constants needed to evaluate Doppler, xenon and burnup reactivity effects are input for each fuel type.

The spatial mesh used in the representation of the 560-fuel assembly Oyster Creek core consists of an array of cubic nodes; one node for each bundle in the horizontal plane and 24 axial nodes in the axial direction. The GPUN input model is consistent with the calculational features of NODE-B and is acceptable.

An important segment of the NODE-B input model is the data used for the normalization of the results to measured data. Appropriate selection of top and bottom albedos, reflector constants and partial fuel factors allows the user to minimize the deviations between measured and calculated data and improve the quality of the input model. These input normalization data sets are constant throughout the analysis of the current and future cycles.

3.2 Verification of Methodology

The methodology employed in NODE-B has been verified by comparing results of calculations with measured data obtained during the operation of Oyster Creek. Both cold zero power and hot operating conditions were included in the verification process. In addition, the performance of the code was verified against measured data from Hatch 1 Cycle 1 operation including Traversing Incore Probe (TIP) data and gamma scan measurements obtained at the end of Cycle 1.

Cold Reactivity

Data from Oyster Creek startup tests at the beginning of Cycles 8, 9 and 10 were used in the verification of the NODE-B cold model. The cold critical tests conducted during the startups were all local criticals. A total of 13 cold critical experiments were conducted during the startups of Cycles 8, 9 and 10. For each local critical configuration, criticality was achieved with both a positive and negative period. Calculations with the NODE-B code yielded a combined average critical k-effective based on the three Oyster Creek cycles, corrected for period and temperature, of 1.002 with a standard deviation of 0.3%.

These results indicate that the cold NODE-B model is capable of calculating shutdown margins within about 0.2% with a standard deviation of about 0.3%. The cold model of NODE-B is found acceptable for application to cold critical experiments of the Oyster Creek cycles which are similar in fuel loading to Cycles 8, 9 and 10.

Hot Reactivity

Core follow calculations were performed for Oyster Creek Cycles 8 and 9 as well as for the Hatch 1 Cycle 1 core. In each of the two Oyster Creek cycles twelve statepoints were calculated. With the exception of a few statepoints in Cycle 9, the core power was at or near the rated level. The mean k-effective for both Oyster Creek cycles was 0.986 with a standard deviation of about 0.2%.

Calculations of seventeen statepoints spanning the entire length of Hatch 1 Cycle 1 resulted in an average k-effective of 0.985 with a standard deviation of about 0.5%. The larger standard deviation of the Hatch k-effective may be due to the plugging of the lower grid plate.

It is seen that in both the Oyster Creek and Hatch verifications NODE-B underpredicts the core reactivity by about 1.5%, with a standard deviation of about 0.5%.

Power Distribution Uncertainties

A measure of the accuracy of the calculated power distribution is provided by the comparison of measured TIP distributions with NODE-B-predicted TIPs. GPU Nuclear's comparisons of these data were made for each of the 12 statepoints spanning Cycle 8 and again for each of the twelve statepoints spanning Cycle 9. In addition to these Oyster Creek comparisons, the NODE-B model was verified against TIP and gamma scan measurements from Hatch 1 Cycle 1.

The verification from the Oyster Creek TIP data leads to a nodal uncertainty of 7.65%. Verification of the NODE-B model against the Hatch 1 Cycle 1 measured TIP data results in a nodal uncertainty of 9.14%. Comparisons with Hatch 1 end-of-cycle 1 gamma scan measurements yield a nodal uncertainty of 7.95%. These results indicate that based on a data base derived from the operation of two Oyster Creek cycles and one Hatch cycle, NODE-B calculates nodal power distributions to within 9.14%. It is expected, therefore, that in core related analyses involving nodal powers, GPU Nuclear will include an uncertainty of 9.14%.

3.3 Methodology Uncertainties

In order to test the validity of the NODE-B model, operating data from two reactors were used including about forty operating states. These states provide an adequate data base for determining NODE-B uncertainties in predicting power distributions and hot and cold reactivities. Based on the calculation-to-measurement comparisons for these states, it is concluded that GPU Nuclear NODE-B predictions of cold reactivity are accurate to within 0.5% with a standard deviation of 0.3%, the hot reactivity predictions are accurate to within 1.5% with a standard deviation of 0.5%, and the nodal power predictions are accurate to within 9%.

Based on the review of the NODE-B methodology and on the verification of the code's ability to reproduce measured data, it is concluded that the code represents an acceptable methodology for performing three-dimensional steady state BWR reload calculations for the Oyster Creek core, that suitable comparisons to operating data were made and that there is a satisfactory agreement between the calculation results and the measurements, and that GPUN has therefore demonstrated an acceptable ability to use the code in cases in which the fuel loading and operating conditions are similar to those of Oyster Creek Cycles 8 and 9.

4.0 CONCLUSIONS

The staff, with the assistance of consultants from Brookhaven National Laboratory, has reviewed the GPUN topical report TR-021, Revision 0, submitted by GPUN to describe and justify the methodology to be used in licensing calculations involving steady state BWR core characteristics. The review evaluated the methodology and the ability of GPUN to use the methodology. Based on this review we conclude that the CPM code as used by GPUN is acceptable for applicable BWR licensing calculations.

5.0 REFERENCES

1. Letter from R. F. Wilson, GPU Nuclear, to J. A. Zwolinski, NRC, March 25, 1986, "Oyster Creek .... Reload Topical Report."

2. EPRI-NODE-B, Advanced Recycle Methodology Program System Documentation Part II, Chapter 17, Rev. 0, September 1977.

3. EPRI-THERM-B, Advanced Recycle Methodology Program System Documentation Part II, Chapter 17, Rev. 0, September 1977.

4. ARMP: Advanced Recycle Methodology Program Systems Documentation, Part I, Chapter 1, Rev. 0, September 1977.

5. BWR Hybrid Power Shape Monitoring System EPRI NP 3195-CCM Volumes 1 and 2, February 1981 and Volume 3, September 1983.

6. D. L. Delp, D. L. Fischer, J. W. Harriman, and M. J. Steadwell, "FLARE - A Three-Dimensional Boiling Water Reactor Simulator," GEAP-4598, July 1964.

7. G. S. Lellouche and B. A. Zolotar, "Mechanistic Model for Predicting Two-Phase Void Fraction for Water in Vertical Tubes, Channels and Rod Bundles," EPRI-NP-2246-SR, February 1982.

8. Letter from R. F. Wilson, GPU Nuclear, to NRC, April 16, 1987, "Oyster Creek .... Reload Topical Report 021."

9. Letter from R. F. Wilson, GPU Nuclear, to NRC, July 10, 1987, "Oyster Creek .... Reload Topical Report 021."

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