UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION REPORT BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

PHYSICS METHODOLOGY

BIG ROCK POINT PLANT

DOCKET NO. 50-155

Consumers Power Company, by letter dated October 30, 1978, submitted for staff review, a description of the reactor physics analysis methods employed at the Big Rock Point Plant. Additional information was provided in letters dated June 15, 1981 and December 15, 1981 in response to staff requests. Finally, a revised report describing the methods was submitted in April, 1982. The Reactor Physics Section of the Core Performance Branch has reviewed these submittals and has prepared the following evaluation:

1. Description of Report

The revised report consists of a section describing the Big Rock Point reactor with emphasis on its differences from modern boiling water reactors, a summary description of core models and calculational techniques employed for steady state physics calculations, and a verification of the results against experiments and higher order calculations.

1.1 Description of Core

Big Rock Point is rated at 240 megawatts thermal and contains 84 fuel assemblies and 32 cruciform control rods. The 16 outer control rods are strong (all poison tubes filled with B_4C) and are used only for shutdown. The 16 inner rods are weaker (40 inner poison tubes per rod are empty). Typically eight or ten of the weak rods are initially in the core at power and are used for long-term reactivity control. BRP fuel is similar to standard BWR fuel except that the assemblies are

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larger (11 x 11 rod array) and since all the interassembly water gaps are the same size, the assemblies are internally octant symmetric. Current BRP reload fuel has 113 fuel rods (composed of three enrichment zones), four gadolinia bearing fuel rods, and four inert rods for improving LOCA performance. Approximately 1/4 of the core is refueled on annual cycles. Although BRP is fundamentally similar to present day BWRs, there are some significant differences in design and mode of operation:

1. The BRP reactor core is small, the active region being about six feet in height and six feet in diameter. An advantage of this is a very stable, leakage controlled power distribution as compared to modern plants whose core volume is about eight times larger. To compensate for the high leakage associated with the small core, reactivity (K-infinity) and, hence, fuel enrichment must be higher than for most later plants.
2. Although shorter, BRP fuel assemblies, are wider than modern plants (7½" pitch versus 6"). The BRP 11 x 11 assembly is roughly the same in rod diameter and pitch to the modern 8 x 8 BWR assembly. Because of the larger assembly, the ratio of control rods to interior assemblies is one to two rather than the typical one to four, i.e. a "D" lattice.
3. BRP has external recirculation loops with constant velocity pumps, therefore flow control is not employed, and maneuvering is done entirely with control rods. This is a disadvantage as far as plant flexibility, but greatly simplifies predictive physics analysis and power distribution surveillance.
4. BRP has only 32 control rods, as opposed to around 200 in the large modern plants. Since the reactivity inventory is about the same as a larger plant, individual control rod worths are generally larger for BRP. During operation, banking the control rods in groups of greater than two rods would result in unacceptable axial power shapes, so that X - Y symmetry is limited to half core rotational, rather than quadrant or octant.

5. LPRMs are present, but they are not part of the reactor protective system. A high flux trip is provided by three excore detectors.
6. In-core power distribution measurements are provided by the activation of flux wires, rather than a movable TIP detector. There are only eight measurement locations arranged in four symmetrically located pairs. These are employed to verify calculated axial power shapes, but because of the small number of locations, they are not considered useful for radial power measurements.
7. The primary coolant system is pressurized to 1350 psi versus the typical 1000 psi. Maximum exit void fractions are about 55%, which is much lower than modern plants.
8. There is no on-line power distribution monitoring system comparable to later plants. The LPRMs are used to monitor changes in power distribution, but there is not an on-line thermal margin calculation.

1.2 Core Physics Analyses

The main calculational tool for the calculation of the steady state core physics characteristics of Big Rock Point is the GROK code. This is a three-dimensional nodal code which is based on the FLARE code and is similar to other such codes used by the industry. It is used to calculate power distributions, margin to thermal limits, control rod and rod notch worths, reactivity coefficients, incremental burnup and overall core reactivity. The core is modeled in nine axial segments. Each node consists of one axial segment of a fuel assembly and its surrounding water and is roughly a cube (7½ x 7½ x 8 inches).

The GROK code requires that the values of k-infinity, M^2 and the ratio of power to flux (kappa/nu) be provided for each node in the core as a function of local conditions. These quantities are precalculated by the CASMO code and entered into GROK in the forms of coefficients in fitting equations. CASMO is a multigroup two-dimensional integral transport theory code for calculation of the neutronics parameters of BWR and PWR lattices. It is capable of explicitly representing the design features of BWR lattices including water gaps, cruciform control rods, gadolinia burnable poison and incore instrument channels. It contains a "built-in" cross-section library and requires, as input, only the lattice description. This code performs the same function as the EPRI CPM code and give essentially identical results.

CASMO calculations are performed for each different type of fuel assembly for uncontrolled, partially controlled, and completely controlled conditions. Assembly parameters are obtained as a function of power level (fuel temperature), void content, burnup, void dependent burnup, burnable poison content, xenon concentration and samarium concentration. Auxiliary codes are used to reduce the results of these calculations to polynomial form for use as input to GROK.

Core thermal-hydraulic conditions are computed in GROK by a state-of-the-art procedure that is employed in other nodal codes. The EPRI void-enthalpy correlation which accounts for sub-cooled boiling is used. The assembly pin peaking factor algorithm used in GROK combines a beginning of life local peaking factor obtained from the CASMO infinite lattice calculation with the gross horizontal power tilt across the assembly and an axial peaking factor obtained by a spline fit to the nodal axial power shape for the assembly. The horizontal power tilt is obtained by a quadratic fit to the nodal power in the assembly and its immediate neighbors.

Core thermal-hydraulic limits, linear heat generation rate (LHGR), minimum critical heat flux ratio (MCHFR) and minimum critical power ratio (MCPR) are calculated. The Hench-Levy correlation is used for MCHFR and the Exxon Nuclear XN2 correlation is used for MCPR.

The uncertainties in the various parameters calculated by the GROK/CASMO code combination have been determined by comparisons with measured data or higher order calculations. Radial assembly power distribution uncertainties were obtained relative to fine mesh diffusion theory (PDQ7) calculations. Comparisons were made for rodded BOC cores and unrodded EOC cores at 0, 25, and 50 percent average voids. The same CASMO calculations were used to provide input to both codes. The standard deviation in the comparison was 2.29 percent leading to a 95/95 value of 3.77 percent (one-sided tolerance).

Axial power distributions were compared to measured flux wire profiles from the preceeding five cycles of the Big Rock Point reactor. After correction of GROK results for spacer effects, the 95/95 value of the uncertainty is 5.30 percent.

Since Big Rock Point does not have an incore monitoring system or a process computer, the CASMO/GROK code system is used for core follow as well as design and analysis. Accordingly, the uncertainty in the determination of core thermal-hydraulic limits has been determined.

The uncertainty on the peak heat flux is a combination of radial, axial, and local peaking factor uncertainties. The local peaking factor calculated by GROK was compared to that calculated by PDQ7 to obtain its uncertainty value. The PDQ7 calculation had one mesh point per fuel pin so that pin powers were available. The pin peaking factor comparisons showed that the 95/95 value of the uncertainty was 5.65 percent. An additional component in the local peaking factor uncertainty is the CASMO/PDQ7 code combination. Comparison with local power distribution measurements has shown that the 95/95 value of this uncertainty is 3.19 percent. Combining the radial, axial, local and CASMO/PDQ7 uncertainties with the two percent heat balance uncertainty yields a total 95/95 value of the peak heat flux uncertainty of 9.40 percent.

Uncertainty in the APLHGR (for comparison to MAPLHGR limits) is obtained by combining by means of a propagation of errors analysis, the uncertainty in the nodal power and the uncertainty in the nodal exposure (which determines the MAPLHGR limit). This results in a value of 7.84 percent for this uncertainty.

A similar propagation of errors is carried out to determine the uncertainty analysis to be ascribed to the value of MCHFR. This is determined to be 0.3228. That is, the calculated critical heat flux ratio must be greater than 3.3228 to assure that the minimum critical heat flux ratio of 3.0 is met.

The uncertainty in the void coefficient determination was obtained from a statistical propagation of errors calculation assuming that the void fraction was a function of core flow, inlet enthalpy and core pressure (for a particular heat input). Uncertainties were obtained for these quantities. They were then combined with the uncertainty in the void model to obtain the total uncertainty. The uncertainty in the k-effective due to voids was obtained from analysis of coastdown data for the last five cycles of Big Rock Point. The final uncertainty is the void coefficient is calculated to be 3.5 percent.

2. Summary of Evaluation

The following discussion summarizes our evaluation of the Big Rock Point physics methodology report.

The codes used by the licensee are similar to those used throughout the industry. Sufficent description of the codes is presented in the report, or by reference to previously approved reports, to support the conclusion that they are state-of-the-art and are acceptable.

Qualification of nodal codes by comparison with higher order calculations is an acceptable procedure. Since the methodology is to be used only for the analysis of the Big Rock Point plant, its verification against Big Rock Point data is acceptable. The uncertainty values obtained for bundle power peaking, axial power peaking and local peaking are consistent with those obtained by similar methods and are acceptable.

Since the code system is also used for core monitoring, uncertainty values have been derived for the core thermal limits, linear heat generation rate and critical heat flux ratio. Minimum critical power ratio is also used as a thermal limit but the uncertainties are included in the correlation which determines the safety limit and are not routinely monitored for this purpose. These uncertainties are obtained from the basic uncertainties described above by statistical combination with other thermal-hydraulic measurement uncertainties. This is an acceptable procedure.

Sufficient data are included in the report to support the conclusion that the values given for the various uncertainties are reasonable and acceptable.

3. Evaluation Procedure

The review of the report on Big Rock Point Methodology was conducted within the guidelines provided in Section 4.3 of the Standard Review Plan. Sufficient information is provided to permit a knowledgeable person to conclude that the methods and procedures are state-of-the-art and are acceptable. This conclusion is stengthened by the fact that comparisons with measurements have shown that acceptable results are produced.

4. Regulatory Position

On the basis of our review, which is described above, we conclude that the "Big Rock Point Physics Methodology Report," Revision 1, dated April 15, 1982, is acceptable for reference in licensing actions relating to the Big Rock Point Nuclear Plant. It is acceptable for calculating the parameters listed in the attached table, and we further conclude that the uncertainties given in the table are acceptable.

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Big Rock Point should continue to monitor the performance of these codes and to further refine the various uncertainties. The staff wishes to be kept informed of any significant development.

Table

Physics Parameters and Uncertainties (95/95)

Parameter	Uncertainty
Bundle Power	3.77 percent
Axial Power	5.30 percent
Local Peaking	5.65 percent
Peak Heat Flux	9.40 percent
MAPLHGR	7.80 percent
MCHFR	.3228 units
Void Coefficient	3.51 percent

5. Acknowledgements

This evaluation was prepared by W. Brooks.

Date