Consumers Power Company

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June 15, 1981

Director, Nuclear Reactor Regulation
Att Mr Dennis M Crutchfield, Chief
Operating Reactors Branch No 5
U S Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 -
BIG ROCK POINT PLANT - REACTOR PHYSICS METHODOLOGY

NRC letter dated April 24, 1981 requested additional information as described in its enclosure, pertaining to Big Rock Point reactor physics methodology. The majority of the questions involve the uncertainties in calculated and measured core parameters.

The application of uncertainties in Big Rock Point reactor physics methodology is based on common practice, engineering judgment and available operating data. No rigorous analysis exists to support fully the currently used uncertainties. Therefore, a benchmarking effort has been initiated to provide analytical justification for these uncertainties. The major part of this new effort will be verification of calculated power distributions and peaking factors and the uncertainties associated with flux wire measurements. The existance of little or no benchmark-quality data results in the need to perform higher order calculations for much of the verification work.

The need to perform additional analysis to address the requested information was brought to the attention of the NRC Project Manager - Big Rock Point. It was agreed that Consumers Power Company should provide as much information as possible within the 45-day response period and provide an estimate of when a full response to all of the requested information could be submitted. Therefore, Attachment 1 to this letter provides our current responses to the information request and it is estimated that a full response can be provided by December 15, 1981.

Gregory C Withrow (Signed)

Gregory C Withrow
Senior Licensing Engineer

CC JGKeppler, USNRC
NRC Resident Inspector - Big Rock Point

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ATTACHMENT 1

Responses to the Enclosure of NRC Letter dated April 24, 1981
Reactor Physics Methodology

A first reading of the document describing the Big Rock Point Physics Methodology leads to the conclusion that the GROK code, with input preparation from CASMO, PDQ-7 and other standard cross section preparation codes, is used for both reload design and analysis and as a replacement for or adjunct to the process computer. When used for reload applications the code calculates power distributions, critical rod patterns at zero power and at operating conditions, rod sequences and rod worths within the sequence, reactivity coefficients (void and Doppler), power distributions and margins to thermal limits. For core follow (process computer) applications the code is used to calculate margins to thermal limits, segment burnups, and flux wire activation profiles (for comparison with measurements).

1. Please verify, clarify, or modify the above understanding, as appropriate.

Response

This conclusion is accurate.

2. For those parameters which are to be used as input to safety analyses please provide the following:

   a. The accuracy with which the parameter is calculated including the supporting data. Is the comparison with experiment, with higher order calculation, or other?

   b. Comparison of results of GROK with those of the code used to calculate the parameter for the FSAR analyses or the currently used reference analysis for each event.

   c. Indicate whether the nominal value is used in the safety analysis or is the uncertainty added before comparison with safety analysis value?

Response

The parameters used as input to the safety analysis include:

Beta effective
Prompt neutron lifetime
Void coefficient
Doppler coeffecient
Scram reactivity curve
Margin to fuel damage during slow rod withdrawal (essentially a complete analysis)

a. Analysis is underway to estimate the accuracy of these parameters. Results will be supplied later.

b. The following table shows a comparison of values computed by GROK for Cycle 17 and the values used in the Final Hazards Summary Report.

Parameter	FHSR	GROK, Cycle 17
Doppler Coef ($\Delta k/k$/% power)	-5.42×10^{-5}	-10.67×10^{-5}
Void Coef ($\Delta k/k$/unit void)	BOC -.20 EOC -.10	-.1453 -.1050
Beta effective	.007	.00577
Prompt neutron lifetime (seconds)	40×10^{-6}	37.3×10^{-6}

A comparison of the scram curve that GROK computed for Cycle 17 and the reference analysis that was submitted June 20, 1974 (Docket 50-155, License DPR-6, Proposed Technical Specifications Change) is given by Figure 1 attached. The dropped rod worth that yields 280 cal/gm energy deposition was determined to be 1.8% $\Delta k/k$. The maximum in sequence dropped worth for Cycle 17 was calculated to be 0.47% $\Delta k/k$.

c. It is our current practice to compare best estimate values to those used in the safety analysis with no uncertainty applied. However, for the transient event which typically determines fuel operating limits (ie, turbine trip without bypass valve operation) the void coefficient is the most critical neutron kinetics parameter. The results of this accident become more severe with a larger negative void coefficient. Considerable margin exists between the Cycle 17 BOC void coefficient as predicted by GROK and the value used in the FHSR.

3. A brief description of the role GROK plays as a process computer surrogate. What parameters does it calculate? How are inputs obtained (ie, is there a direct connection from reactor to computer or is GROK used only on an off-line basis)? What is the accuracy of the calculated quantities? Are nominal values used to compare with operating limits (ie, the limits themselves take into account the uncertainties) or are uncertainties added before comparing?

Response

As a process computer surrogate, GROK is used off-line to either update fuel exposure or to evaluate thermal margin and compute flux wire activation profiles. There is no direct connection from the reactor to GROK; all inputs must be done by hand. To evaluate thermal margin, the inputs are taken from a hand heat balance and consist of power level, inlet subcooling, core flow and control rod pattern. The quantities computed by GROK are:

heat flux
MAPLHGR
MCHFR
MCPR
assembly power

The GROK calculation includes a 4% uncertainty factor that is applied to heat flux, MCHFR and MCPR that accounts for fuel densification and mechanical variations in the fuel. Also, a 10% uncertainty factor is applied to enthalpy increase as the coolant flows up the channel and is input to the MCHFR and MCPR calculations.

Before the computed values of heat flux, MAPLHGR and MCHFR are compared to the operating limits, a fluxwire correction factor is applied. The fluxwire correction factor is derived by comparing the GROK computed fluxwire shapes to the measured shapes and finding the largest difference between the two sets of curves. The fluxwire measurement is taken at the same time the heat balance is made. The single largest difference is then applied to the computed thermal limits (heat flux, MAPLHGR & MCHFR) for the entire core. In addition, 2% is applied to these limits in addition to the fluxwire correction factor. The fluxwire correction is not applied to bundle power, however, 1% uncertainty is applied. An analysis to determine the uncertainty in the calculated parameters will be performed.

4. With respect to the fluxwire measurements please provide estimates of the accuracy of these measurements for axial power distribution in a single four assembly cluster and for making radial power distribution determinations.

Response

The accuracy of the fluxwire system for measuring axial power distributions will be analyzed. There are only eight fluxwire holes, and four of these are symmetric to the other four, so that the system is generally regarded as inadequate for measuring radial power distribution.

5. Provide information with respect to initial critical configurations in terms of Δk as well as the number of notches.

Response

For Cycle 14 (Figure 4-1 of the Methodology Report) $\Delta k/k = .00160$. For Cycle 15 (Figure 4-2) $\Delta k/k = -.00003$.

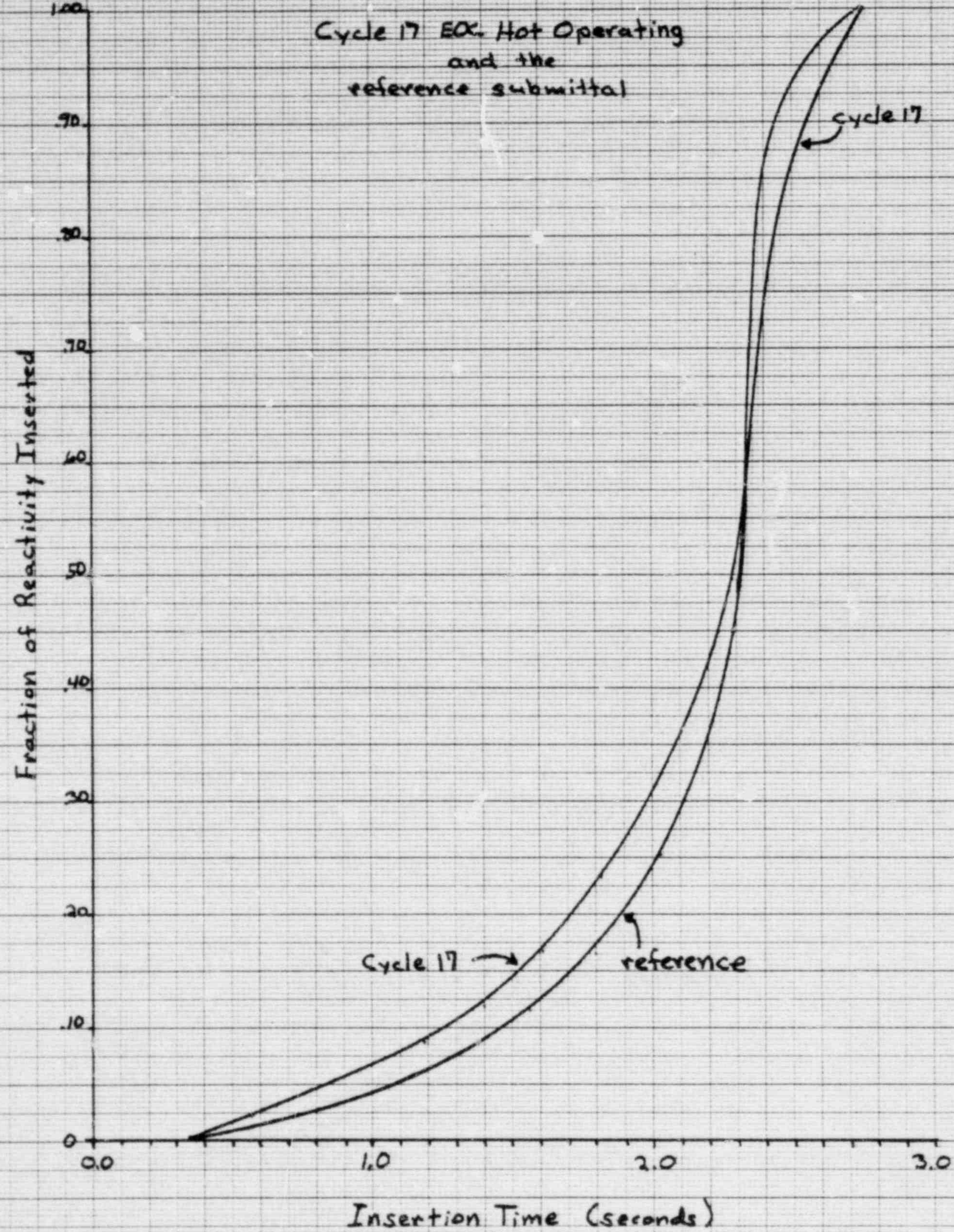
6. The fluxwire - GROK comparison data in the report should be summarized to provide calculational uncertainties for radial and axial peaking factors and any other power distribution factors used in safety analyses or core follow.

Response

An analysis to provide these uncertainties will be performed.

Figure 1

Scram Function Comparison



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