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ENCLOSURE Responses to Request for Addl Info Regarding Exxon Report XN-75-27, Suppl #1...

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NRC FORM 195 (2-76)

CP&L

Carolina Power & Light Company

December 2, 1976

FILE: NG-3514(R)

SERIAL: NG-76-1554

Director of Nuclear Reactor Regulation
ATTN: Robert W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2
DOCKET NO. 50-261
FACILITY OPERATING LICENSE NO. DPR-23
RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING EXXON REPORT XN-75-27, SUPPLEMENT 1

Dear Mr. Reid:

By letter of November 22, 1976, you requested that Carolina Power & Light Company (CP&L) provide responses to the request for additional information regarding Exxon Report, "Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors," XN-75-27, Supplement 1. Enclosed are the responses to these questions.

It is our understanding that this enclosure provides the Staff with the information necessary to complete the review of Exxon Report XN-75-27, Supplement 1, and complete the assessment concerning the H. B. Robinson reload report XN-76-39, "H. B. Robinson Unit 2 Cycle 5 Reload Safety Analysis Report."

Yours very truly,

E. E. Utley
Vice President
Bulk Power Supply

MFP/dkm

Enclosure

12251

12251

336 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION REGARDING EXXON REPORT XN-75-27, SUPPLEMENT 1

Received w/Ltr Dated 12-2-76

A. Power Distribution Calculations (Chapter 3)

QUESTION 1

To what extent does the empirical correction to the baffle cross sections used in PDQ calculations depend on reactor state variables such as: control rod pattern, core exposure and distribution, core power level and distribution, fuel loading pattern, core flow, etc.?

RESPONSE

The empirical correction to the baffle cross sections is used in PDQ calculations on H. B. Robinson Unit 2 only. It consists of adjusting the fast group diffusion coefficient to match transport theory calculations or the core/baffle interface. This particular model has been checked out to H. B. Robinson Unit 2 Cycles 2 and 4. The comparisons between measured and calculated can be found in XN-75-27, Supplement 1. With respect to various reactor state variables, the following reemphasis is stated:

Control Rod Patterns - The ability to predict control rod worths provides an indirect check on whether the model predicts power distributions correctly. In other words, if the control rod worth methodology is correct, the rod worths can still be in error if the power distributions are in error. XN-75-27, Supplement 1 indicates no bias with respect to different control rod configurations. The empirical "baffle" model is primarily a power distribution effect.

Core Exposure and Distribution - The correction has been demonstrated to be independent of core exposure and distribution; see Table 3.B in the report and attached update of this table. The regionwise and assemblywise standard deviations throughout Cycle 4 indicate the same consistent deviations (1-2%).

Power Distributions - Referring again to the attached table and the standard deviations and noting that the power distributions toward the end of the cycle were considerably flatter than those in the beginning of the cycle. Additional confirmation is formed in the Cycle 2 comparisons summarized in Table 3.9 of XN-75-27, Supplement 1.

Loading Patterns - The empirical "baffle" model has been found to apply to both Cycles 2 and 4 of H. B. Robinson Unit 2, and additional confirmation will be provided once Cycle 5 starts up. It is noted that the Cycle 5 loading pattern has been analyzed using both the empirical and the analytical models and both were found to give acceptable power distributions for the start of the cycle.

Core Power Level and Core Flow - Sensitivity to these parameters have not been looked at yet; however, these will be addressed generically in a future supplement to XN-75-27.

QUESTION 2

The peripheral bundles having two faces to the reflector have been suggested as a cause of the radial bias between measured and calculated assembly powers. In order to quantify this effect, have any 2-D transport or other calculations been performed to determine the ability of diffusion theory to represent the reflector in this case?

RESPONSE

The effect of peripheral assemblies with two faces toward the reflector has been looked at with a two-dimensional transport theory model. The purpose of this calculation was to see if the albedo near a corner was significantly different from albedoes distant from a corner. The results of the calculation indicated that the albedoes did not differ significantly; thus, this analysis was not pursued any further.

QUESTION 3

What spatial and spectral averaging procedures were used in determining the baffle, shroud and reflector cross sections? What reflector temperature and distribution (if any) were used? How sensitive are the peripheral assembly powers to this representation?

RESPONSE

The basic broad-group cross sections for the baffle are generated using a fuel spectrum. The cross sections are subsequently input to a 1G transport theory calculation in which two fuel assemblies, the baffle, and the water reflector are described. An identical diffusion theory calculation is then mocked up and the fast diffusion coefficient adjusted until the diffusion theory albedo at the baffle/fuel interface matches that of transport theory. The resulting cross section set is then used in 2G diffusion theory calculations. Reflector cross sections are obtained by running a XPOSE calculation with water in all the regions and the temperature at about 550°F, which corresponds to the core inlet temperature of the coolant. Communications with various plant staff have indicated that the ΔT axially in the reflector is small and is therefore neglected. The assembly powers on the periphery are not very sensitive to the reflector representation. This is because the leakage correction at the boundary is all taken at the fuel/baffle interface.

QUESTION 4

If the radial bias is due to an instrument failure or an approximation in the INCORE software, the "measured" powers will be incorrect and low in the center of the core. If the PDQ calculations are then lowered in the center to force agreement with "measurement," will the INCORE results reflect this adjustment and hence will "measured" powers be incorrectly reduced in the center, further increasing (rather than eliminating) the measurement error? What tests have been made to determine instrumentation failure? Have the fixed and movable incore detector signals been compared? How is detector depletion and nonlinearity accounted for?

With regard to INCORE software, the flux detector signal/power correlation used to determine the "measured" assembly powers is expected to have significant exposure dependence due to assembly power peaking in the neighborhood of the instrument tube. Due to the higher exposure in the center of the core (relative to the periphery), the neglect of this effect could result in a decrease in the

measured powers in the center of the core and a radial bias as observed. Therefore, define in detail the dependence of the signal/power ratio used in INCORE on the local assembly variables (control, exposure, moderator temperature, etc.) and its basis.

RESPONSE

Adjustment of the PDQ calculational results to reduce or remove the radial bias determined during Cycle 4 startup measurements will have no appreciable effect on the INCORE results. The PDQ information that is used in INCORE is employed to determine peak rod power in both instrumented and noninstrumented assemblies and for extrapolation from instrumented assembly locations to define the assembly and peak rod power levels in uninstrumented assembly locations. It has very little effect on the resulting gross radial power distribution produced by the INCORE calculation, and this effect can be essentially eliminated by "folding" all the instrumented locations back into a quarter core representation.

In the course of the reanalysis of the H. B. Robinson Unit 2 Cycle 4 startup data, raw reactor flux traces were compared directly to PDQ calculations. These comparisons were made to eliminate any possibility that the reactor flux map processing code INCORE, utilized by the H. B. Robinson staff, was introducing spurious data, which later came to be called measured.

The investigation process consisted of hand normalizing the data which is input into the INCORE code to the common detector location. A reactor core detector signal map for only the measured locations was then generated. From PDQ, a similar map of the calculated detector responses was obtained.

Comparison of the measured data which had not been handled by INCORE to the PDQ calculated data showed no discrepancies of a systematic nature. There were no "tilts" or biases between fresh and burned fuel. The only other significant observation was that some of the raw measured data from symmetric locations had signal differences of greater than 3% in very early (in the cycle) flux maps. As the cycle progressed, these variations dropped to well under 3%.

The conclusion of the investigation was that the output from the INCORE code should be used for direct comparison to the PDQ calculations.

Additional studies performed by Westinghouse and quoted in recent SAR's verify that up to 25% of the detector locations can be inoperable with no appreciable effect on either gross power distribution or calculated peaking factors. Thus, correction of the calculational bias in PDQ will not adversely affect the radial power distribution calculated by INCORE.

The calibration of the excore detectors to the movable incore detectors is performed periodically by procedure and is performed to assure accurate axial and radial representation of the core behavior by the excore detectors. Four strings of fixed incore detectors are located in the Robinson core as an experimental program and are not employed in the power distribution measurement programs.

With regard to the INCORE software, it is the responsibility of ENC to supply:

1. Assemblywise radial power distributions as a function of cycle exposure.

2. Instrumented assembly thimble fluxes for above power distributions.

3. Assembly local peaking factors.

4. U-235 fission cross section for the detector.

The exposure dependence of the flux detector signal/power correlation is accounted for by the fact that the described input data to INCORE is generated as a function of cycle exposure. The same holds true for the rodded plane. It is also pointed out that there are no fixed detectors employed in either H. B. Robinson Unit 2 or D. C. Cook Unit 1, and the movable detectors are used in several different assemblies with different exposures. All the movable detectors are intercalibrated by passing them all through the same instrumented location. It is, therefore, felt that a radial bias could not result from the concerns stated in this question.

QUESTION 5

What procedure was employed to average over the axial dimension in the 2-D PDQ calculation and what effect does this approximation have on the observed radial bias?

RESPONSE

No axial averaging is performed in the 2-D PDQ calculations. This is justified on the basis of the fact that the cores run essentially rod free. Significant axial effects due to power changes and control rod motions are calculated using the 3-D XTC model.

QUESTION 6

Does an axial bias exist between PDQ and the measured assembly power distribution, e.g., a PDQ overprediction in unrodded axial regions and underprediction in a rodded region?

RESPONSE

No axial bias between the PDQ calculations and the measured assembly powers has been observed. It is noted the incore maps are processed using two basic data decks; namely, one for the unrodded plane and one for the rodded plane. The code (INCORE) processes the flux map data using both decks and interpolates between the rodded and the unrodded regions.

QUESTION 7 IS NOT APPLICABLE TO H. B. ROBINSON.

B. Procedure for Determining Delayed Neutron Parameters (Chapter 5)

QUESTIONS

1. Derive or explain in detail the basis for equation (5.2).
2. Derive equations (5.3) and (5.4).

RESPONSE

The equations under discussion are:

$$5.2 \qquad \bar{I} = k_{delayed}/k_{prompt}$$

$$5.3 \qquad \bar{\beta}_g = \bar{I} \frac{\sum_n \phi_n \sum_i N_i \nu_{i,n} \sigma_{fi,n} \beta^g_{i,n}}{\sum_n \phi_n \sum_i N_i \nu_{i,n} \sigma_{fi,n}}$$

$$5.4 \qquad \bar{\lambda}_g = \frac{\sum_n \phi_n \sum_i \lambda^g_{i,n} \beta^g_{i,n} \nu_{i,n} \sigma_{fi,n} N_i}{\bar{\beta}_g \sum_n \phi_n \sum_i \nu_{i,n} \sigma_{fi,n} N_i}$$

The equations as written above are correct, but differ slightly from those appearing in XN-75-27, Supplement 1, due to typographical errors. In equation 5.3, the subscript on $\bar{\beta}$ appears as a "q" rather than a "g" as above. In equation 5.4, the nuclide concentration N_i was omitted in the equation appearing in the supplement. Equation 5.2 is correct as it appears.

The importance factor as defined in equation 5.2 represents the relative effect of delayed versus prompt neutrons. The delayed neutrons are emitted at lower energies than the prompt neutrons and therefore see a different total cross section. The ratio of $k_{effective}$ for delayed and prompt neutrons represents their relative probability to produce neutrons, thus the term importance factor. In equation 5.1:

$$5.1 \qquad \rho = \frac{1^*}{T} + \bar{I} \sum_{i=1}^{6} \frac{\beta i}{1 + \lambda i\, T} ,$$

both prompt and delayed neutrons are represented. The importance factor is used when summing the reactivity contributions from the prompt and delayed neutrons.

Equation 5.3 is a weighted average of the individual $\beta_{i,n}$ for each group to determine a single $\bar{\beta}_{i,n}$ representative of the core. The weighting factor is the number of neutrons produced from the fissioning nuclides.

Equation 5.4 is the weighted average of the individual $\lambda_{i,n}$ to produce a single $\bar{\lambda}$ for the core. The weighting factor is the number of delayed neutrons produced by the fissioning nuclides.

FLUX MAP SUMMARY

Map No.	MWD/MT	D Bank	Core Avg AO	Core Avg FZ	$F_{\Delta H}^{N}$	KW/ft	Max. Peaking Values F_Q	F_R	F_{XY}	F_Z	Standard Deviation 4C	5	6	7
209	567	209	- 0.54	1.117	1.503	10.13	1.764	1.277	1.479	1.170	0.50	2.24	2.24	2.19
210	1060	212	- 0.49	1.112	1.499	10.15	1.768	1.280	1.481	1.160	3.71	1.77	1.78	1.70
211	1278	215	- 1.18	1.109	1.501	10.23	1.781	1.283	1.490	1.160	3.78	1.82	2.12	1.83
212	1716	215	- 1.36	1.108	1.503	10.22	1.780	1.288	1.496	1.170	3.30	2.20	2.70	2.76
213	2151	219	- 1.30	1.103	1.483	9.98	1.737	1.267	1.497	1.160	3.47	1.64	1.92	1.82
214	2493	219	- 3.27	1.123	1.486	10.19	1.774	1.253	1.473	1.170	4.24	1.90	2.58	2.69
215	2525	219	0.95	1.112	1.479	9.90	1.723	1.246	1.471	1.200	3.15	2.24	2.61	2.96
216	2525	219	- 4.09	1.136	1.511	10.24	1.782	1.273	1.486	1.180	3.73	1.83	1.87	2.5[illegible]
217	2588	220	- 1.73	1.103	1.471	9.91	1.725	1.239	1.451	1.150	2.53	1.64	1.85	2.04
218	2740	190	-13.75	1.260	1.478	11.15	1.941	1.246	1.441	1.290	0.72	1.56	1.32	1.57
219	3012	212	- 2.83	1.114	1.459	9.81	1.708	1.232	1.431	1.150	2.36	1.27	1.36	1.28
220	3627	209	- 3.97	1.121	1.448	9.79	1.703	1.226	1.442	1.150	0.93	1.18	1.11	1.29
221	4058	211	- 2.85	1.113	1.449	9.69	1.688	1.232	1.410	1.140	0.99	1.52	1.31	1.74
222	4438	212	- 3.51	1.116	1.435	9.64	1.677	1.219	1.405	1.140	1.38	1.28	1.38	1.41
223	5434	215	- 3.37	1.114	1.405	9.45	1.645	1.202	1.365	1.140	0.10	1.01	1.02	1.10
224	5902	217	- 2.91	1.106	1.394	9.35	1.629	1.202	1.364	1.140	0.55	1.09	1.00	1.16
225	6210	206	- 4.32	1.122	1.395	9.42	1.641	1.204	1.356	1.160	0.79	1.40	1.21	1.29
226	6210	198	- 8.67	1.185	1.396	9.82	1.710	1.193	1.348	1.240	1.17	1.50	1.26	1.33
227	6210	191	-14.03	1.273	1.397	10.49	1.828	1.201	1.353	1.340	1.13	1.58	1.32	1.31
228*	6210	218	4.06	1.167	1.401	9.66	1.683	1.200	1.353	1.220	0.51	1.53	1.37	1.43
229	6925	213	- 3.00	1.111	1.379	9.14	1.598	1.195	1.344	1.140	1.15	1.28	1.17	1.30
230	7330	211	- 3.24	1.120	1.375	9.18	1.598	1.140	1.330	1.160	1.56	1.37	1.17	1.47
231	7560	213	- 3.23	1.110	1.362	9.04	1.574	1.194	1.321	1.140	1.74	1.35	1.18	1.30
232	8381	212	- 3.56	1.121	1.354	9.18	1.598	1.194	1.313	1.160	2.31	1.65	1.49	1.5[illegible]
233	8788	213	- 2.89	1.113	1.355	9.03	1.574	1.201	1.319	1.140	1.94	1.54	1.40	1.5[illegible]
234	9469	211	- 3.88	1.131	1.345	9.17	1.597	1.204	1.308	1.170	0.61	1.55	1.40	1.60

All maps taken at 100% power (2200 MWT).

F_Q Includes 1.03 engineering factor and 1.05 measurement uncertainty.

$F_{\Delta H}^{N}$ Includes 1.04 measurement uncertainty.

KW/ft Includes 1.03 engineering factor and 1.05 measurement uncertainty.

* 7-14-76