ENCLOSURE

Evaluation of Report MSS-NA1-P

Report Number:	MSS-NA1-P
Report Title:	Qualification of Reactor Physics Methods for Application to PWRs in the Middle South Utilities System
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Originating Organization:	Middle South Services, Inc.
Reviewed By:	Reactor Physics Section Core Performance Branch Division of Systems Integration Office of Nuclear Reactor Regulation

The Middle South Services, Inc. has submitted a technical report describing the techniques and the supporting data for the reload design methodology for the PWRs of the Middle South Utilities System (ANO-1 and ANO-2). The information of the subject report has been supplemented and revised (in some sections) with the submittal of responses to staff review questions dated October 16, 1981. The report includes four major sections which deal with the computational model, its uncertainties, and applications to reactor operations and safety evaluations. The described methodology is based on the ARMP system of codes which has been benchmarked on ANO-1 data and limited data from ANO-2.

1. Summary of the Report

The Calculational Model is based on the Advanced Recycle Methodology Program (ARMP) developed by EPRI (1). Nuclide concentrations (including depletion and fission product chain data) and micro- and macro-scopic cross section data tables are produced by the spectrum code EPRI-CELL. Depletion is computed with the diffusion based two-dimensional programs PDQ7 and HARMONY (2,3). Lumped absorber data for PDQ7/HARMONY are generated by a capture fraction matching procedure between PDQ7 and EPRI-CELL for burnable poisons or CPM (the Collision Probabilty Module segment of ARMP) for the control rods. The PDQ7/HARMONY is run for core-wide solutions, nodal code normalization, local peaking factors and assembly loading patterns. The EPRI-CELL uses the GAM code for slowing down calculations, and the THERMOS for thermalization calculations. CINDER is used for fission products and trans-uranics. A resonance treatment is used to tabulate group-wise resonance integrals as functions of temperature and potential scattering.

The procedures used to calculate core safety margins are the same as those in the model benchmarking assuring consistency of procedures in benchmarking and analysis. Data used for the benchmarking are from ANO-1 cycles 1, 2 and 3 and some data from ANO-2.

2. Summary of Evaluation

We have reviewed the sections of MSS-NA1-P which refer to verification and reliability applications to reactor operations and applications to safety evaluations.

Model Verification and Reliability is assured by quantification of the reliability factors to be used in safety-related calculations. In this report the term "reliability factor" is used to describe the allowances to be used to assure conservatism. The reliability factor is always larger than the 1σ uncertainty. A bias factor is also applied on the average difference between the measured and calculated value of a parameter. The reliability factors applicable to the main physics parameters are listed in Table 1.

In the following, we shall review the verification and uncertainty of important physics parameters.

- o Rod Worth and Soluble Boron Coefficient uncertainty depends on boron titration, rod calculational worth and boron coefficient calculational worth uncertainties. The statistical sum of all of the above is close to the titration uncertainty; hence, it is concluded that the other two are negligibly small.

- Temperature Coefficient uncertainty was determined using pooled data from isothermal temperature coefficient measurements from ANO-1 and ANO-2. The data pooled were properly qualified. The overall reliability factor was found to be 4.0 pcm/°F.

- Doppler and Power Defect uncertainty could not be measured from the available ANO-1 data. A conservative 10% reliability factor for the Doppler coefficient and 20% for the power defect have been accepted.

- Isotopic composition uncertainties have been computed with CPM (Collision Probability Module, a high order transport code) and the results compared to measured values from Yankee Rowe and Saxton (4,5). CPM is part of the ARMP package of codes but is not a production code. The EPRI-CELL has been adjusted to yield comparable results with CPM. Calculated and measured comparisons of atom densities and plutonium isotope ratios are very good.

- Delayed Neutron Parameters i.e., β_{eff} and λ uncertainties consist of several components which depend on experimental values of β and λ, spatial distribution of nuclides and the flux average values of β. It is conservatively estimated that 3% is the overall uncertainty factor for both β_{eff} and the core average neutron lifetime l^*.

- Power Distribution Uncertainties are divided into nodal and local power distribution uncertainties. For both ANO-1 and ANO-2 power distribution reliability factors are computed by Rh signal detector power measurement and corresponding local power calculation. There are 364 Rh detectors in ANO-1 in the instrument hole of 52 fuel assemblies arranged in seven axial core levels. There are 44 assemblies similarly instrumented in ANO-2 in five axial core levels. A power distribution is constructed using the Rh measured values to which the grid flux depressions are superimposed. Grid flux depressions raise the peaking factors by 1-2%. Calculated values are obtained using fine mesh diffusion methods. The methods are such as to assure computational consistency of instrument signal normalization and power distribution calculations. A total of 28 core maps from ANO-1 over three cycles were used for the comparison of measured and calculated values. Analysis of these results indicate good agreement. A statistical analysis model is described. Simulation errors due to unsteady conditions in the core before or during the map measurements are accounted for. Likewise the detector intercalibration effect is included in the estimation of the uncertainty. For each detector level, an average axial bias is determined. Finally, a statistical estimate is made of all model uncertainty contributions.

The local power distribution uncertainty is associated with the PDQ calculation of the peak/average assembly power. The evaulation includes: first, a comparison of the CPM to experimental data; next, a comparison of PDQ to CPM; and finally, a combination of the above to establish the uncertainty of PDQ predictions with respect to experimental data. Four experiments from the KRITZ critical facility were used to benchmark CPM. For locations next to water holes, the average CPM error is -.7% and the standard deviation 1.9%. The PDQ vs CPM comparison was made for typical ANO-1 fuel over a wide range of burnup. The MSS calculations are consistent with the PDQ vs CPM comparisons described in the ARMP documentation. In general, it demonstrated that PDQ overpredicts peak pin power relative to CPM. This overprediction is proportional to the magnitude of the pin power. This is in agreement with the behavior of PDQ vs CPM found in the ARMP documentation. A conservative value for the pin factor was assumed. The overall pin calculation factors using the ARMP model were:

pin factor uncertainty = .020
pin factor bias = .005

Power Distribution Reliability Factors include local linear heat generation peaking factors F_Q^N and the enthalpy rise hot channel peaking factor $F_{\Delta H}$. The F_Q^N includes the effects of the nodal model, i.e. axial flux gradient and grid effects and the local peaking factor, both discussed in preceding paragraphs. A combined upper bound reliability factor of 10% at the 95% probability/95% confidence level has been estimated. Likewise ARMP methods are used for $F_{\Delta H}$ and involve nodal uncertainties in the calculation of the assembly average power sharing and the PDQ pin peaking F_p. A combined 95/95% uncertainty factor of .052 is computed. A conservative value of .057 for use in both ANO-1 and ANO-2 has been accepted.

Applications to Reactor Operations are used to illustrate the predictive and monitoring functions of the ARMP methodology. These applications are not intended to define the procedures, rather to illustrate the calculations.

- Startup Physics Tests are performed after each refueling and the results are compared to model predictions. The predicted value of a parameter is the difference of the ARMP computed value minus the bias.

- Power distribution measurements are performed by interpreting in-core flux detector signals using ARMP MSS generated constants for the calculations. When ARMP data are used for monitoring peaking factors limited by technical specifications, allowance is made for the effect of the uncertainty on the inferred power distribution due to the uncertainties in the data. Such uncertainties have been computed and tabulated. In addition, allowances are made for uncertainties which may be required by technical specifications, administrative procedures for fuel engineering and materials tolerances, etc.

- Isotopic inventory uncertainty depends on the accuracy of the computation of the spatial burnup distribution and on the uncertainty of the local inventory calculation as a function of nodal exposure. Both have been dealt with in preceeding paragraphs.

Applications to Safety Evaluation Calculations are described in this paragraph as an illustration of such applications, rather than as a systematic demonstration of safety evaluations.

- Nuclear Heat Flux Hot Channel Factors F_Q^N are computed using the three dimensional power distribution obtained from the nodal code, supplemented with the local peak to assembly power ratios obtained via a PDQ solution. The F_Q^N is then obtained as follows:

  $$F_Q^N = (1 + \text{Reliability Factor} - \text{Bias}) * F_Q^N(\text{ARMP})$$

- Nuclear Enthalpy Rise Hot Channel Factors $F_{\Delta H}$ are also calculated through the three dimensional nodal solution in conjunction with the local pin to assembly power ratios. The $F_{\Delta H}$ is calculated as follows:

  $$F_{\Delta H} = (1 + \text{Reliability Factor} - \text{Bias}) * F_{\Delta H}\ (\text{ARMP})$$

- Rod Worths are calculated using the three-dimensional nodal power distribution and then varying the rod position while the independent core parameters are held constant. The rod worth are calculated as:

  $$\Delta\rho = (1 \pm \text{Reliability Factor} - \text{Bias}) * \Delta\rho(\text{ARMP})$$

  (The Reliability Factor is added or subtracted depending on the application).

- The Moderator Coefficient measures the change in reactivity due to moderator changes caused by temperature pressure or void variations. The moderator coefficient α_m is calculated as:

  $$\alpha_m = \alpha_m(\text{ARMP}) - \text{Bias} \pm (\text{Reliability Factor})$$

  (The Reliability Factor is either added or subtracted, whichever is appropriate depending upon the application.)

- Fuel Temperature (Doppler) Coefficient α, is calculated through the value of the power coefficient which consists of the moderator and fuel temperature changes. The Doppler coefficient is the remainder after the moderator coefficient is removed from the power coefficient which is computed with a three-dimensional power distribution as a function of power and exposure. α_D is computed as follows:

  $$\alpha_D = \alpha_D \text{ (ARMP) } (1 - \text{Bias} \pm (\text{Reliability Factor}))$$

  (The Reliability Factor is added or subtracted depending upon the particular application).

- Boron Concentration Coefficient (α_B) which measures core reactivity variation as a function of boron concentration is calculated with the aid of the three dimensional nodal power distribution. The value of α_B is computed as follows:

  $$\alpha_B = \alpha_B \text{ (ARMP) } (1 - \text{Bias} \pm (\text{Reliability Factor}))$$

  (The Reliability Factor is added or subtracted depending on the particular application).

- Effective Delayed Neutron Fraction (β_{eff}) is determined by weighting local values from each fissile nuclide for its contribution to local fission as determined by PDQ. The value is also adjusted for fast fission, resonance escape and fast leakage. β_{eff} is computed as follows:

  $$\beta_{eff} = \beta_{eff} \text{ (ARMP) } (1 - (\text{Reliability Factor}) - \text{Bias})$$

- Prompt Neutron Generation Time (1^*) is computed as a function of core exposure from two-dimensional PDQ calculations. Two group flux weighted parameters are used to compute the slowing down and the thermal diffusion time for $1/v$ absorbtion 1^* is given as follows:

  $$1^* = 1^* \text{ (ARMP) } (1 - (\text{Reliability Factor}) - \text{Bias})$$

- Shutdown Margin is computed with a sequence of calculations with the shutdown reactor at zero power and with the highest worth rod stuck out and all other rods inserted. The shutdown margin is adjusted for the reliability factors and biases for rod worth, temperature defect and Doppler defect as discussed elsewhere.

- Scram Worth Versus Time is computed in a two-step sequence. First, measurement data relating rod position to time after-rod release are related to the results of a three-dimensional nodal model of reactivity insertion vs rod position. Reliability factors for rod worth are taken into account.

3. Evaluation Procedure

The report MSS-NA1-P has been reviewed within the guidelines provided by the Standard Review Plan, Section 4.3 and the applicable parts of Section 15, i.e., 15.4.1, 2, 3, 7, and 8. Sufficient information is provided in the report and its additions and revisions to permit a knowledgeable person to ascertain that the methods and techniques used are satisfactory and the data employed are adequate.

4. Regulatory Position

On the basis of our review we concluded that Technical Report MSS-NA1-P may be referenced in licensing actions by the Middle South Services Inc. for the physics calculations for the Middle South Services ANO-1 and ANO-2 Power Stations. We recommend that the Middle South Services Inc. perform periodic re-evaluations of the reload methodology to provide continuing assurance of model applicability.

REFERENCES

1. Advanced Recycle Methodology Program (ARMP) System Documentation, CCM-3 Research Project 118-1, September 1977.
2. Cadwell, W. R., "PDQ-7 Reference Manual", WAPD-TM-678, Westinghouse Electric Corporation (January 1967).
3. Breen, R. J., Marlowe, O. J. and Pfeifer, C. J. "HARMONY: System for Nuclear Reactor Depletion Computation", WAPD-TM-478, Westinghouse Electric Corporation (January 1965).
4. Nodvik, R. J., "Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core 1 Spent Fuel, Including Isotopes of Elements Thorium through Curium", WCAP-6086 (1969).
5. Nodvik, R. J., "Saxton Core II Fuel Performance Evaluation", Part II, WCAP-3385-56.

TABLE 1

RELIABILITY FACTORS FOR ANO-2 BENCHMARK CALCULATIONS

Parameter	Reliability Factor	Bias
F^N_Q	$RF_{FQ} = 0.10$	0
$F_{\Delta H}$	$RF_{F\Delta H} = 0.057$	0
Rod Worth	$RF_{RODS} = 0.05$	0
Temperature Coefficient	$RF_M = 4.0$ PCM/^oF	0
Doppler Coefficient	$RF_D = 0.10$	0
Doppler Defect	$RF_{DD} = 0.20$	0
Boron Worth	$RF_B = 0.05$	0
Delayed Neutron Parameters	$RF_\beta = 0.03$	0
	$RF_{\ell*} = 0.03$	0