UNCERTAINTY EVALUATIONS OF CASMO-3/MASTER SYSTEM FOR PWR CORE NEUTRONICS CALCULATIONS

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ABSTRACT

Uncertainties in core neutronic calculations of CASMO-3/MASTER, which is a KAERI developed core nuclear design code system, were evaluated via comparisons with measured data. Comparisons were performed with plant measurement data from one Westinghouse type and one ABB-CE type plant and two Korean standard type plants. The CASMO-3/MASTER capability and levels of accuracy are concluded to be sufficient for the neutronics design including safety related parameters related with reactivity, power distributions, temperature and power coefficients, inverse boron worth and control bank worth.

1. INTRODUCTION

The neutronics design methodologies of PWR consist of cross section generation and reactor core simulation systems. These methodologies are used for design of initial and reload cores, calculation of parameters for safety analyses, and core follow calculations. This paper describes the performance of a neutronics computer code system CASMO-3/MASTER which is developed as a part of KAERI design system ADONIS.

CASMO-3[1] is a multi-group two-dimensional transport theory code developed by Studsvik for burnup calculations on fuel assemblies or simple pin cells. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with gadolinium, burnable absorber rods, in-core instrument channels, water gaps, and control rods in the regions separating fuel assemblies. It can also handle a model for generation of baffle/ reflector and axial reflector regions. Nuclear data are collected in a library containing microscopic cross section in 70 or 40 energy groups which covers the energy range of 0 to 10 MeV. The effective cross sections for the gadolinium loaded fuel are generated by the MICBURN[2] code.

MASTER[3] is a nuclear design code based on the two group diffusion theory to calculate the steady-state and transient behavior of pressurized water reactor core in three-dimensional Cartesian geometry. Its neutronics model solves the space-time dependent neutron diffusion equations with Nodal Integral Method, Nodal Expansion Method or Finite Difference Method. The transverse leakage model is treated by a parabolic approximation and multi-level coarse mesh rebalancing and asymptotic extrapolation method are implemented to accelerate the convergence of iteration process. MASTER uses microscopic cross sections provided by CASMO-3. It performs depletion calculation using microscopic cross sections and also has the reconstruction capability of pin-by-pin information using Method of Successive Smoothing with Modified Analytic Solution. The neutronics calculation is linked to thermal-hydraulic feedbacks for the fuel temperature, moderator temperature and density.

This paper describes the verification and level of accuracy of the CASMO-3/MASTER computer code system via benchmark comparisons and uncertainty evaluations. The comparisons include reactivity, assembly power distribution, temperature coefficients, power coefficient, and control rod worths. The measurements data were obtained from Yonggwang Unit 1 (YGN-1) cycles 1 through 4, Yonggwang Unit 3 (YGN-3) cycle 1, Yonggwang Unit 4 (YGN-4) cycle 1 and Palo Verde Unit 1 (PVNG-1) cycles 1 through 4. All results are compared with the previously used criteria or uncertainties.

2. BENCHMARK COMPARISONS AND UNCERTAINTY EVALUATIONS

The verification of the CASMO-3/MASTER reactivity predictive capabilities was performed by comparison against core measured critical boron concentration data. The measured data were obtained from movable detector system of YGN-1 and incore instrument system of YGN-3, and PVNG-1. The reactivity differences were estimated by the difference of the HFP critical boron concentration multiplied by boron worth which is calculated at each burnup step. Figure 1 shows the reactivity differences and uncertainty which is characterized by ±322 pcm. Figure 2 shows the HFP critical boron concentration differences and shows that the differences are less than the typical PWR criteria of ±50 ppm except for PVNG-1 cycle 1 case.

The assembly radial power distribution at HFP ARO condition predictive capability was verified by the comparisons of measured and predicted distributions. The cycle maximum root mean squares errors are shown in Table 1. Comparisons were performed at least one data point per 30 EFPD for each cycle, if available. All predictions meet the typical PWR criteria of ±5 % in root mean squares error.

The isothermal temperature coefficient (ITC) is the change in core reactivity resulting from a 1 °C change in moderator and fuel temperatures. The ITC is used herein because it is directly measurable, unlike the moderator temperature coefficient (MTC) which excludes the effects of concurrent fuel temperature change. Since the ITC is the sum of MTC and fuel temperature coefficient (FTC), the uncertainty of ITC can be conservatively used for the MTC. Figure 3 shows the differences between calculated and measured values and the uncertainty in ITC. Each calculation matched the conditions of the respective measurement as closely as possible. As a result the ITC uncertainty is characterized by tolerance limits of ±2.53 pcm/°C on a 95/95 probability/confidence level.

The power coefficient is the change in core reactivity due to a 1% change in core power level. Since the power coefficient is measured by rodded method which keeps the constant core average moderator temperature, it implies Doppler only power coefficient. Therefore, the FTC uncertainty can be evaluated using the power coefficient comparisons. Figure 4 shows the differences between calculated and measured values and the uncertainty in power coefficients. In the uncertainty evaluation the data of YGN-4 cycle 1 95% power level was excluded, since the measured value is highly inconsistent with the YGN-3 data which has the same core design of YGN-4 and the data trend is not physical. The uncertainty of power coefficient is characterized by tolerance limits of ±1.39 pcm/%power on a 95/95 probability/confidence level. The uncertainty of FTC can be evaluated from the power coefficient difference data, which is the uncertainty unit of % relative difference as shown in Figure 5. This translates the FTC tolerance limits of ± 13.82 % on a 95/95 probability/confidence level.

Insertions of control rod banks from start-up physics tests were simulated with three-dimensional MASTER calculations. All MASTER calculations were performed at the conditions of the respective measurements. Figures 6 and 7 show the differences between calculated and measured values and the uncertainty in the individual and total bank worths, respectively. The individual and total bank worths uncertainties are characterized by tolerance limits of ±12.50 % and ±10.05, respectively, on a 95/95 probability/confidence level. Note that the total bank worth uncertainty slightly greater than typical criteria of 10% is because of the lack of measured data. The maximum % difference of total bank worth is only 5.85%.

The inverse boron worth (IBW) is expressed as the number of PPM of soluble boron required to change the core reactivity by 1 pcm. Measured boron worths were obtained as a by-product of the control rod bank measurements performed at start-up physics test as such reflect the errors inherent in the measured critical boron concentrations and measured rod bank worths. Figure 8 shows the % differences between calculated and measured values and the uncertainty in IBW which is characterized by tolerance limits of ±10.12 % on a 95/95 probability/confidence level.

3. CONCLUSIONS

This paper establishes the CASMO-3/MASTER computer code system as adequate tools for nuclear core design and safety-related core neutronics calculations with well quantified uncertainties. Table 2 shows the results of uncertainty evaluations compared with previously used uncertainty and physics test criteria guidelines of ANSI 19.6.1.[4] The plant types include Westinghouse reactor with 157 assemblies, ABB-CE reactor with 217 assemblies and Korean standard reactor with 177 assemblies. The fuel types include 17x17 array with a Westinghouse guide tube configuration and 16x16 array with an ABB-CE guide tube configuration. The capability of the system and levels of accuracy are concluded to be sufficient for the core neutronics design including all safety related parameters.

REFERENCES

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3. C. H. Lee, et. al., "MASTER User's Manual," KAERI/TR-560/95, 1995.
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Table 1. Cycle Maximum Root Mean Squares Errors of Assembly Radial Power Distributions

Plant	Cycle	Max (RMS)
YGN-1	1	1.97 %
	2	2.05 %
	3	1.94 %
YGN-3	1	1.43 %
PVNG-1	1	1.78 %
	2	2.51 %

Table 2. Summary and Comparisons of Biases and Uncertainties

	CASMO-3/MASTER			DIT/ROCS		ANSI
	Bias	KSc	# of Data	Bias	KSc	19.6.1
Reactivity (pcm)	0	322	1488	(1)	370	NA
Power Distribution Ass. Max. RMS (%)	0	Max. 2.51	NA	0	NA	Max. 5
ITC and MTC (pcm/°C)	0	2.53	39	(2)	2.84	3.6
Power Coefficient (pcm/%power)	0	1.39	9	(3)	1.1	NA
FTC (%)	0	13.82	9	4.1	18.1	NA
IBW (%)	0	10.12	20	-2.57	10.95	NA
Bank Worth Individual (%) Total (%)	 0 0	 12.50 10.05	 52 10	 7.5 2.04	 15.5 6.52	 15 % 10 %

DIT/ROCS Bias : (1) Core Average Enrichment and Burnup Dependent

(2) Critical Boron Concentration Dependent

(3) Power Dependent

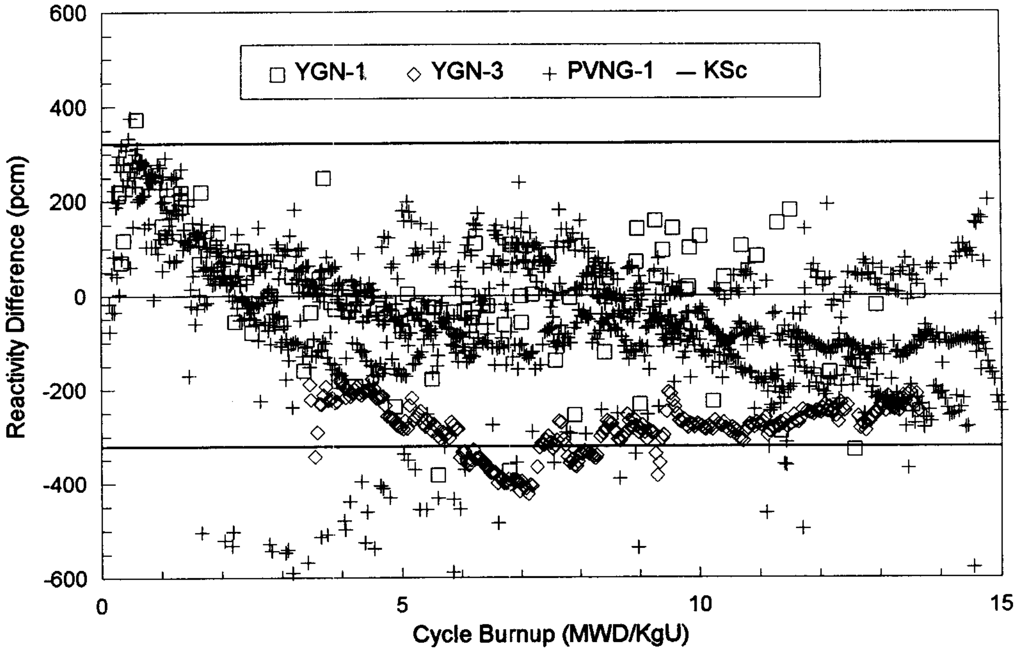

Figure 1. Reactivity Differences and Tolerance Limits

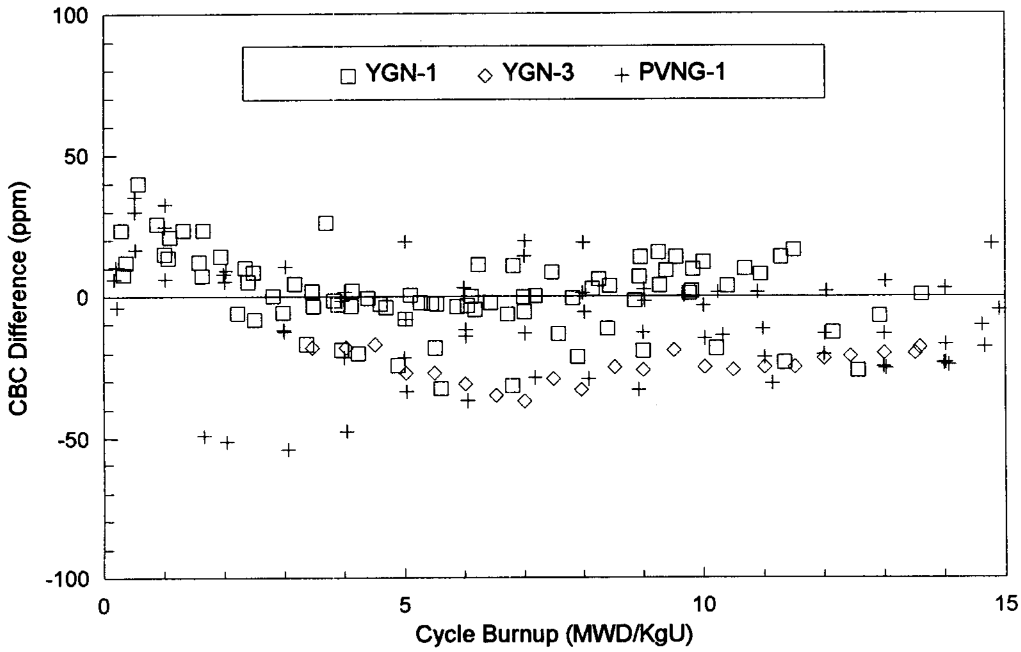

Figure 2. Critical Boron Concentration Differences

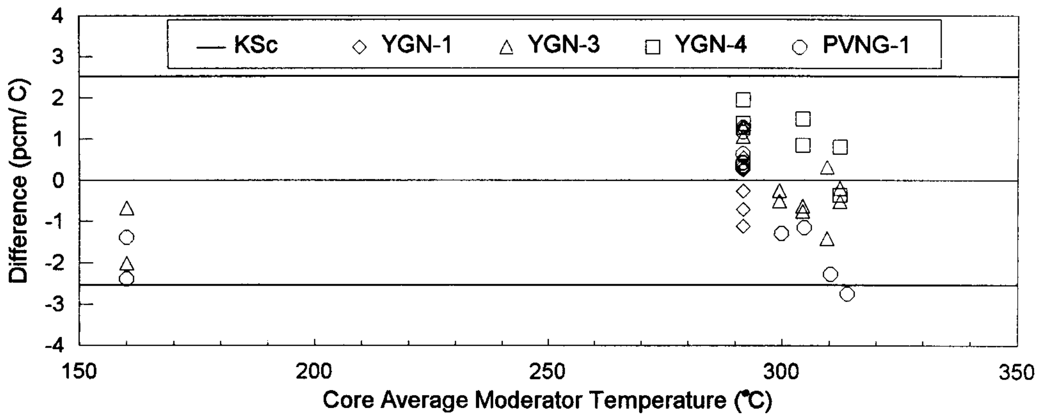

Figure 3. ITC Differences and Tolerance Limits

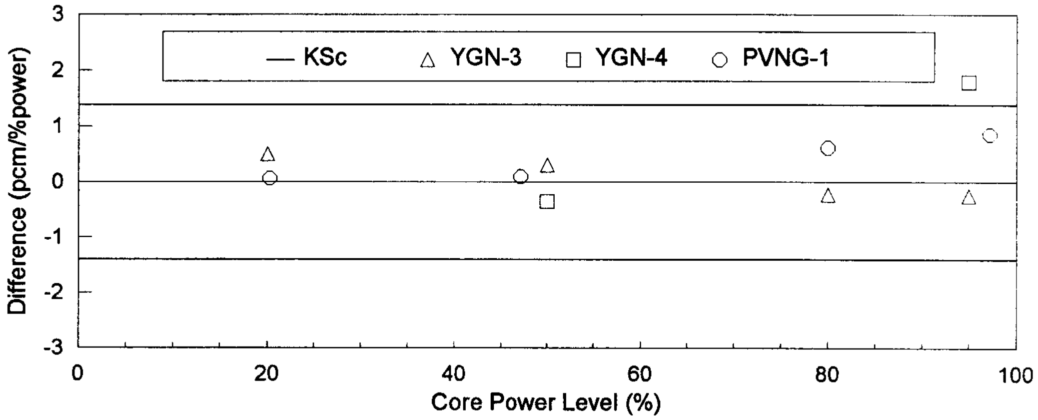

Figure 4. Power Coefficient Differences and Tolerance Limits

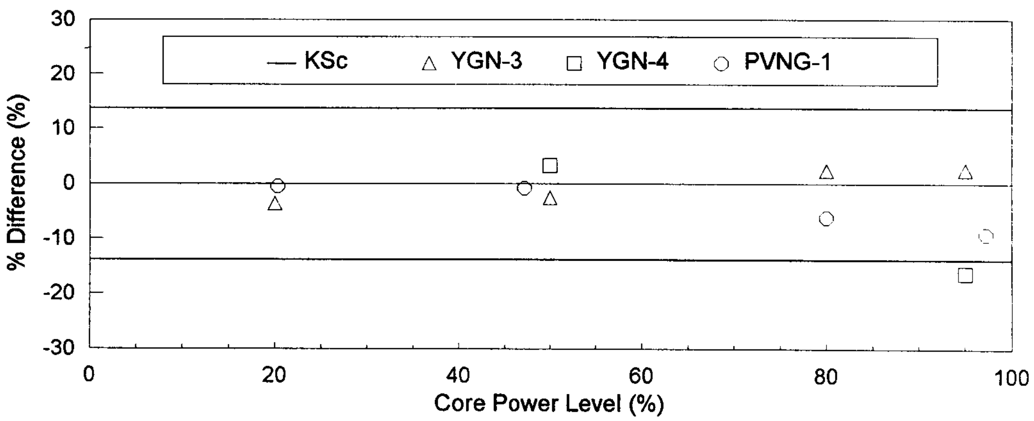

Figure 5. FTC % Differences and Tolerance Limits

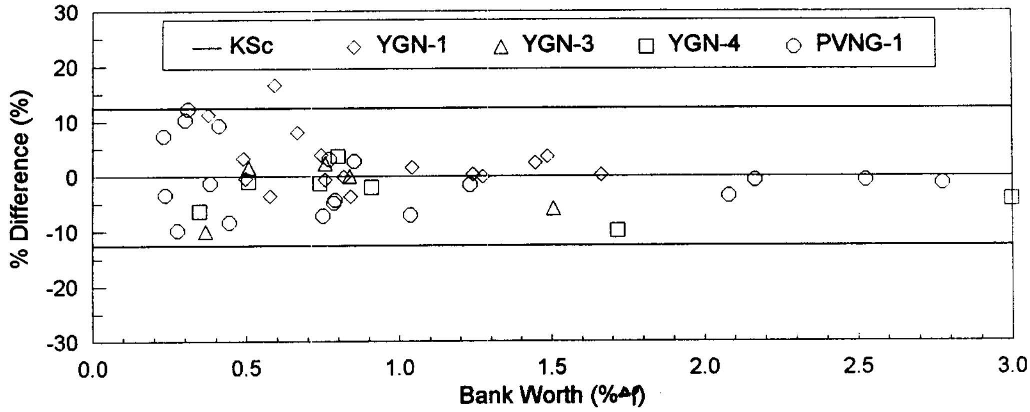

Figure 6. Individual Bank Worth % Differences and Tolerance Limits

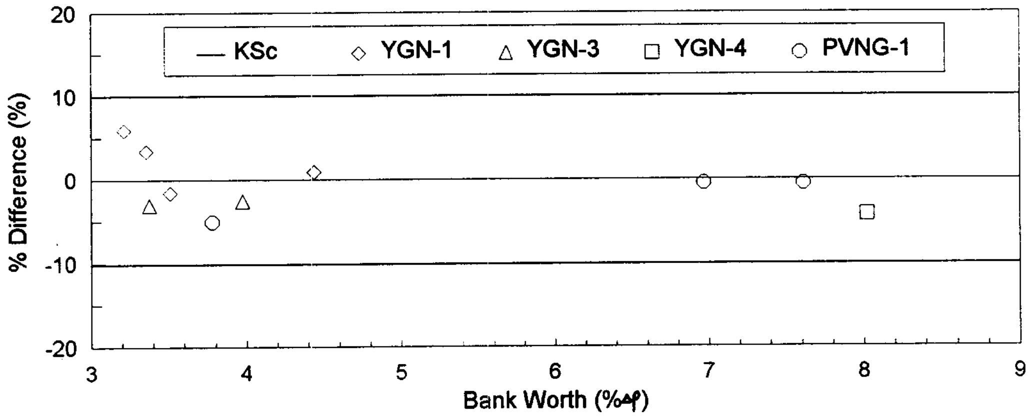

Figure 7. Total Bank Worth % Differences and Tolerance Limits

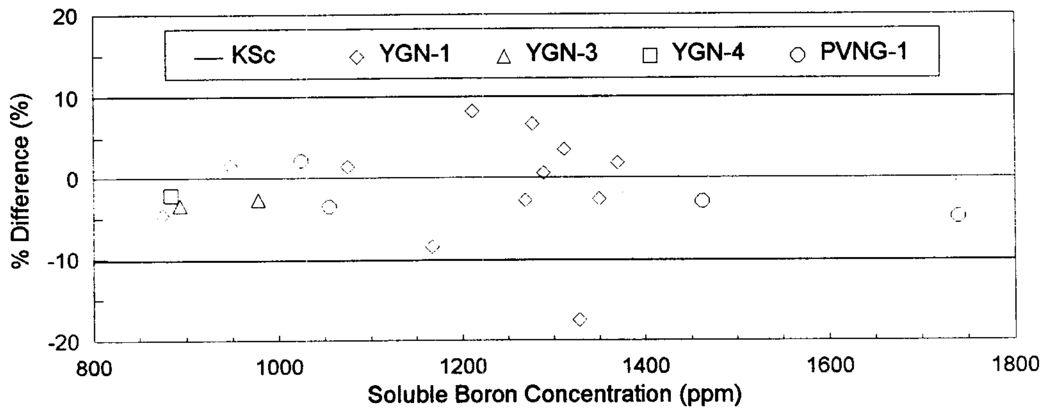

Figure 8. Inverse Boron Worth % Differences and Tolerance Limits