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An Adaptation of the SAV Standard Nuclide Chain for the CASMO3/MEDIUM3 Procedure

Chang Ho Lee and Young Jin Kim
Korea Atomic Energy Research Institute
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CASMO3 / MEDIUM3 계산절차를 위한 SAV의 표준 핵종 연쇄모델의 수정

이창호 · 김영진
한국원자력연구소
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Abstract

The nuclide chain model used in SAV90 has been modified for the CASMO3/MEDIUM3 procedure. Since the default nuclide chain in SAV90, using 21 nuclides, is not sufficient to reproduce the CASMO3 results in the MEDIUM3 calculation, the extended nuclide chain models have been investigated and verified with various types of fuel assemblies. Among the extended nuclide chain models proposed, the 22 nuclide chain model, which contains only Pu238 additionally to the 21 nuclide chain, is recommended in terms of both accuracy and computing efficiency. Using this model core follow calculations for YGN−1 have been performed. The results showed good performance when compared to plant measurements.

요 약

SAV90에서 사용되고 있는 핵종 연쇄모델을 CASMO3 /MEDIUM3 계산절차에 상응되도록 수정하였다. 기존의 핵종 연쇄모델은 21개의 핵종으로 표현되어 있어 CASMO3의 계산결과를 MEDIUM3에서 그대로 구현하는데 충분치 않은 것으로 밝혀졌다. 따라서, 이를 해결하기 위해서 기존의 핵종 연쇄모델을 수정 확장시켰으며, 여기에서 분석된 여러 핵종 연쇄모델들중 21 핵종 연쇄모델에 Pu238만을 더 고려한 22 핵종을 가진 연쇄모델이 정확도와 계산효율을 모두 고려할 때 가장 우수한 것으로 나타났다. 이 모델을 이용하여 영광 1호기의 노심연소계산을 수행하였으며, 이를 주요 노심 측정치와 비교한 결과 잘 일치하는 것으로 나타났다.

1. Introduction

The nodal reactor core analysis system SAV79 has been recently replaced by the SAV90 for PWR design and fuel management calculations by Siemens[1]. The main differences of SAV90 are the improved handling and automation of the whole nuclear calculating system, emphasizing the aspects of visualization and easier quality assurance. However, the approved methodology of SAV79 and its verification have been

kept intact. Thus the calculated results of SAV90 can be directly used to predict reactor design parameters on the basis of SAV79 validation experience[2].

The main constituents of SAV90 are the spectral code system FOXS, the core analysis program MEDIUM3 and the dehomogenization program PINPOW2. The FOXS system consists of the preprocessor NUKLAN and the two spectral codes FASER/MULTIMEDIUM and CASMO3[3]. At present all SAV applications depend exclusively upon FASER/MULTIMEDIUM branch. CASMO3 branch provides an option to generate independent cross section data by a simple switch within FOXS.

CASMO3 is the multigroup 2−dimensional transport theory code for burnup calculations on both PWR and BWR assemblies or simple pin cells and is most widely used throughout the world. Unlike the SAV spectral codes, this program was developed to provide assembly homogenized macroscopic cross sections. The SAV concept requires microscopic cross sections supplied to the nodal code, and the depletion and buildup of the isotopes is recalculated in the nodal calculations of reactor core. Therefore when the interface between CASMO3 and MEDIUM3 is constructed, it is important to fully reflect the SAV concept.

There are two main areas to be considered to accomplish this goal. One is the proper processing of CASMO3 output so as to generate microscopic cross section library, for which we developed TRAFO program. The other is to determine the details of nuclide chain model and associated data to be used for isotope depletion in MEDIUM3. The nuclide chain model can be developed based on how MEDIUM3 0D calculations reproduce the CASMO3 results in terms of assembly reactivity as a function of physical state parameters.

In the previous study[4] it was found that the use of standard nuclide model with CASMO3 cross sections in MEDIUM calculations resulted in large deviations, compared to CASMO3 results, in assembly k_∞ at the burnup higher than about 40 MWD/KgU. This is thought to be caused by the simplified nuclide chain, especially in higher actinides, and the nuclide burnup data which are not consistent with CASMO3. Therefore one of the most important things to set up the CASMO3/MEDIUM3 procedure is to define an appropriate nuclide chain model for it.

The purpose of present work is to develop an extended nuclide chain model best fitted to the CASMO3/MEDIUM3 procedure. To begin with, four candidates of nuclide models were derived based on the nuclide chain and burnup data actually used in CASMO3. A recommended nuclide chain is selected from the validation studies using MEDIUM 0D calculations for various assembly fuel types. Then this model is applied to 3−D core follow calculations for the first 5 cycles of Yonggwang unit 1 (YGN−1), and verified by comparing the calculated and measured design parameters such as critical borons, control rod worths, reactivity coefficients, and reaction rate measurements, etc.

2. Nuclide Chain Model

The MEDIUM3 standard nuclide model is consistent with that of FASER. Both codes use the same nuclide chain and burnup data except fission products treatment. They also employ the identical technique in solving isotope depletion equation. The standard chain displayed in Fig. 1 employs 21 nuclides in which higher actinides such as Am and Cm isotopes are neglected.

On the other hand the nuclide chain used in CASMO3 is more detailed as shown in Fig. 2. It is obvious that the standard nuclide chain does not represent Fig. 2 very well. Besides the yield vectors of important fission products are quite different as shown in Table 1. The combining effects imply that the default chain in MEDIUM3 calculation with CASMO3 cross sections will not reproduce CASMO3 results within a prescribed accuracy at high assembly burnup where higher actinides contribute significant portion of reactions. It becomes evident that the

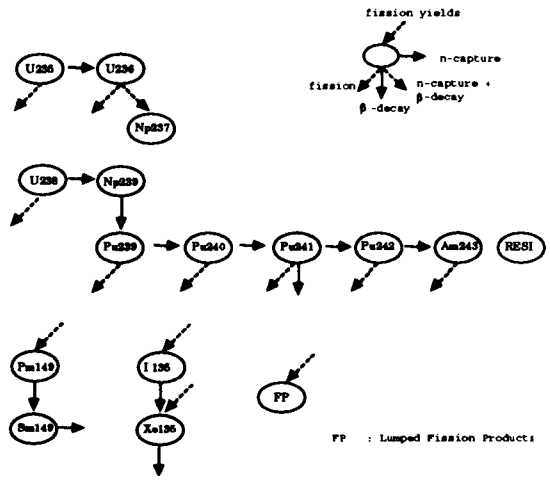

Fig. 1. The Standard Nuclide Chain Used by default in MEDIUM3

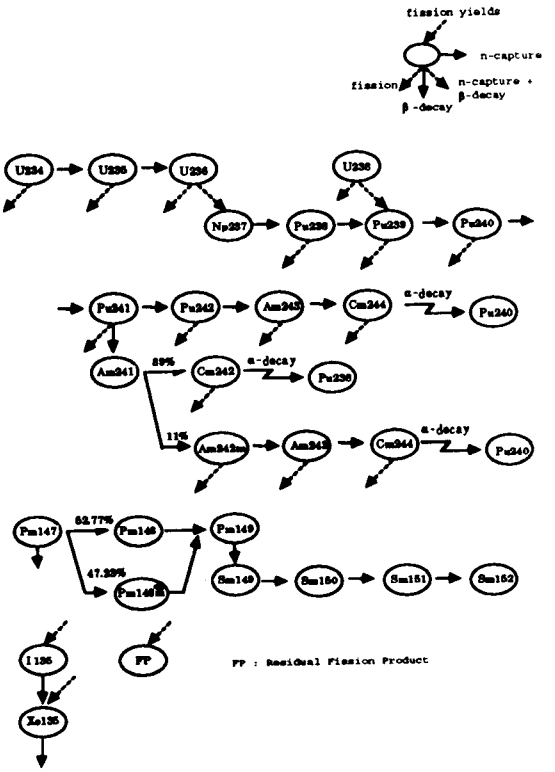

Fig. 2. The Nuclide Chains Used in CASMO3

standard nuclide model should be appropriately adapted for the CASMO3/MEDIUM3 procedure so that it can be applied for entire burnup range without losing accuracy.

As for the accurate reproduction in MEDIUM3, the most promising way is to construct a nuclide chain as closely as possible to that used in CASMO3. It may, however, increase the requirement of computer resources. Therefore a tradeoff may be needed to develop a production version of nuclide chain model. Four candidate chain models, which include 21, 22, 25, and 28 nuclides respectively, are derived from CASMO3 information. Among them the 21 nuclide chain is the same as the SAV90 standard model in most part except the yield vector. In the 22 nuclide chain model, U235 and U238 chains are coupled by adding Pu238 to the 21 nuclide chain. While U235 chain is terminated at Np237 in the 21 nuclide chain model, it is connected to U238 chain through Pu238 produced by n-capture reaction of Np237 in the 22 nuclide one. The 25 nuclide chain describes heavy nuclides as closely to CASMO3 as possible while Sm chain is still approximated. Finally the 28 nuclide chain is constructed in such a way that it can be considered a reference model by which one can expect the exact reproduction of CASMO3. All models are summarized in Table 2.

In generating MEDIUM3 cross section library from CASMO3 output, some cross sections are subjected to correction, for conserving reaction rates, according to the complexity of each nuclide chain. For example, in the cases except the 28 nuclide chain model the effect of Pm149 produced by the decay of Pm147 and Pm148 is included in the fission product(FP) cross section. For the higher actinides which are not specified explicitly, their combined contributions are put in to the special macroscopic cross section RESI, which is additionally defined for the CASMO3/MEDIUM3 procedure. Also, the branch chains in Am241 and Pm147, from which two daughter nuclides are produced, are approximately treated. That is, the major branch is

Table 1. Yield Fractions of Important Fission Products

Nuclide	U-235	U-238	Pu-239	Pu-241	Code
I-135	0.06298	0.06827	0.06447	0.07068	CASMO3
	0.0610	0.060	0.0573	0.058	FASER3
Xe-135	0.00242	0.00028	0.01152	0.00231	CASMO3
	0.0033	0.0023	0.0148	0.0150	FASER3
Pm-149	0.01067	0.01608	0.01239	0.01524	CASMO3
	0.0104	0.0180	0.0130	0.0149	FASER3

Table 2. Candidate Nuclide Chain Models

Model	Heavy Nuclide	Fission Product	Burnable Poison	Structural Meterial
21	Standard(Fig. 1)	Standard(Fig. 1)	BP	B10, H20, STRM, MAC
22	Standard + Pu238	Standard	BP	B10, H20, STRM, MAC
25	Standard + Pu238 Am241, Am242, Cm242	Standard	BP	B10, H20, STRM, MAC
28	Standard + Pu238, Am241, Am242, Cm242	Standard + Pm147, Pm148 + Pm148m	BP	B10, H20, STRM, MAC

handled as normal capture reaction and the other minor one is treated as (n, 2n) since two simultaneous capture processes cannot be specified in the current structure of MEDIUM3 cross section library.

3. Evaluation of Nuclide Chain Model

For the validation check of the extended nuclide chains, a reference has been established in the first place. CASMO3 calculations with very fine time steps produce reference $k\infty$ as a function of burnup and feedback state parameters. Fine time steps assured the avoidance of uncertainty that might come from the use of solution technique with coarse depletion time steps. Also the effect of thermal cutoff energy between 0.625 eV and 1.855 eV has been investigated. Since the cutoff energy has turned out to be insensitive to the results, 1.855 eV is used in this study.

Prior to evaluating the performance of four candidate nuclide models, a sensitivity study has been carried out to eliminate uncertainties which can be caused by inconsistency between CASMO3 and MEDIUM3 0D calculations. First, the decay constants and yield data taken from the CASMO3 information were tested to make sure their correctness. Second it was found that the treatment of α-decay caused instability in the depletion calculation when the recommended time step, 30 EFPD, was used in MEDIUM3 calculations. This instability could be removed with the time step reduced to 15 EFPD.

The extended nuclide chain models were evaluated for typical fuel types loaded in PWRs: normal uranium fuel, 16 WABA poisoned uranium fuel, 8 Gd poisoned uranium fuel, and MOX fuel. The MEDIUM3 0D calculations with CASMO3 library

were performed for the fuel types mentioned above and the results are compared with CASMO3 reference solutions. The scope of comparison included $k\infty$ at reference physical states, various reactivity coefficients, and control rod worth. In the course of evaluation a model was considered successful if the difference was less than 0.1% Δk for the entire burnup domain.

The comparisons of $k\infty$ at reference physical states are shown in Fig. 3. The 21 nuclide model produces very poor results at high burnup especially for uranium fuels, whereas the other models are in good agreement within a difference of 0.1% Δk for all fuel types. It is interesting to note that coupling of U235 and U238 chains by the addition of Pu238, as shown in the result of the 22 nuclide model, is the key for the accuracy. In larger than 22 nuclide chains the increase of the number of nuclides did not improve the accuracy any further within the acceptance criterion desired. Therefore the 22 nuclide model was chosen and subjected to further validation.

The 22 nuclide model is applied to the calculations of reactivity coefficients and control rod worth for all fuel types. Fig. 4 shows the comparison result for the reactivity worths of critical boron, fuel temperature, moderator temperature and moderator density. The results indicate very good agreement. The control rod worth was also reproduced within the criterion for the entire burnup domain.

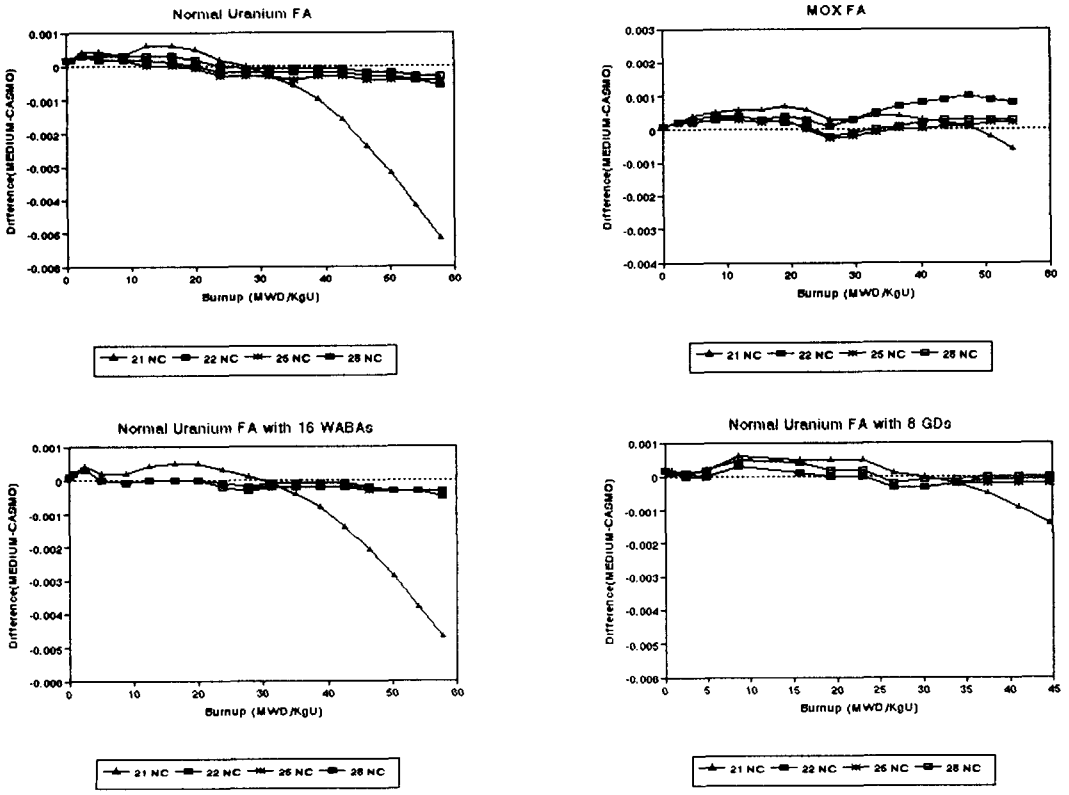

Fig. 3. Comparison of $k\infty$ between CASMO3 and MEDIUM3 0D

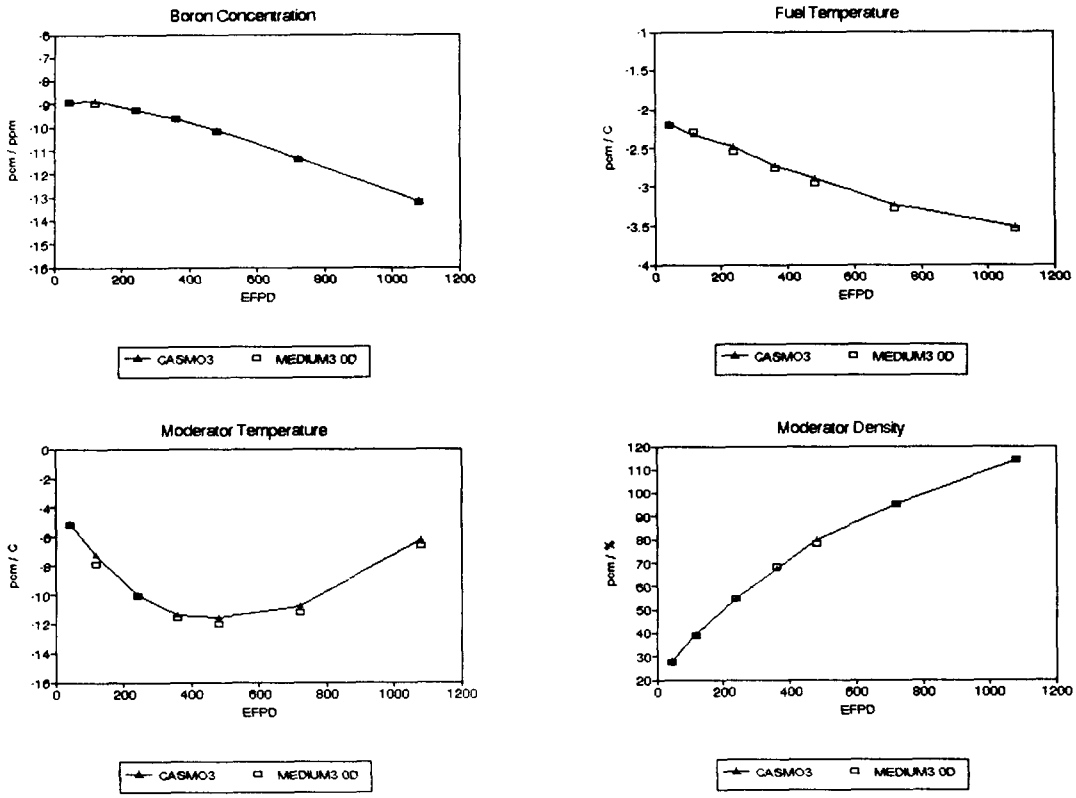

Fig. 4. Comparison of Reactivity Coefficients between CASMO3 and MEDIUM3 0D in the Normal Uranium Fuel Assembly

4. Application of the Extended Nuclide Chain to Core Follow Calculations

Using the 22 nuclide model, the CASMO3/MEDIUM3/PINPOW2 calculations have been performed for cycles 1 to 5 of YGN−1 and compared with the plant measurement data. YGN−1 is a typical Westinghouse 3−loop PWR with 2,775 MWth. The core design consists of 157 fuel assemblies with 17x17 fuel rod array and has the history of various types of fuel loading. Fuel enrichments range from 1.6% to 3.5% with wet annular burnable absorber (WABA) or Gadolinia poisoned rod. The loading schemes are from the typical checker-board pattern for the first cycle to a low leakage pattern featuring loading some fresh fuels inboard. Table 3 shows the summary of core descriptions from cycle 1 to cycle 5.

Cross section libraries and form functions were produced for the fuel types listed in Table 3 by carrying out FOXS runs with CASMO option. The reflector cross sections were obtained by the same method as the SAV standard procedure[5].

In the SAV procedure, control rods are treated according to the DELTA - SIGMA($\Delta\Sigma$) concept in which the presence of control rods is described by adding burnup-dependent $\Delta\Sigma$ to the Σ of the unrodded assembly. The CASMO3 cross section is defined in this study as:

$$\Delta\Sigma = \frac{\Sigma_c}{f_c} - \frac{\Sigma_u}{f_u}$$

Table 3. Core Descriptions of Cycles 1 to 5 of YGN 1

Cycle	Region	No. of FA	Enrichment (w/o)	No. of BPs	Types of Poisoned FA
1	1	53	1.60		
	2	52	2.41	416(WABA)	4, 8, 12
	3	52	3.10	104(WABA)	4, 9
2	1	1	1.60		
	2	52	2.41		
	3	52	3.10		
	3S	4	3.10	64(WABA)	16
	4	48	3.16	416(WABA)	8, 12, 16
3	2	1	2.41		
	3	52	3.10		
	3S	4	3.10		
	4	48	3.16		
	5	52	3.20	368(WABA)	12, 16
4	3	1	3.10		
	4	48	3.16		
	5	52	3.20		
	6	56	3.21	128(PYREX)	8
				144(WABA)	8, 12
5	5	49	3.20		
	6	56	3.21		
	7	52	3.50	112(GD)	4, 8

where f is the discontinuity factor and the subscripts c and u mean rodded and unrodded states, respectively. This formulae is derived from the simplified equivalence theory[6] which offers a consistent frame for homogenizing strongly absorbing assemblies.

All the core follow calculations with the CASMO3 cross sections are based on MEDIUM3 three-dimensional quarter core model. In this model, one node per fuel assembly is used in the radial direction and 16 axial layers for the axial core height. The burnup-corrected NEM option is employed to assure the high accuracy of coarse mesh nodal calculation with only one node per assembly. Reaction rate distributions are calculated by using PINPOW2 which performs dehomogenization for the reconstruction of an accurate local pinwise flux and power distributions.

The measured data from startup physics test and reactor operation provide practical basis for the verification of CASMO3/MEDIUM3/PINPOW2 procedure based on the recommended 22 nuclide model. The startup physics test was simulated at BOC hot zero power conditions for each reload core of YGN−1, and the calculated results were compared with the measured nuclear parameters such as end point critical boron concentration, control rod worths, isothermal temperature coefficients, etc. The reactor operation data compared are critical soluble boron concentrations and reaction rate distributions from movable detector readings at selected assembly positions as a function of cycle burnup.

5. Results of Core Follow Calculations

5.1. Startup Physics Prediction

These tests are performed at hot zero power conditions prior to resuming normal operation for each reload cycle. Three types of measurement data are available:

1. critical boron concentrations with all rods out, control rod banks inserted
2. isothermal temperature coefficients under the same conditions as above
3. reactivity worth of control rod banks.

The critical boron concentrations measured at the startup of each cycle provide information to confirm the reactivity loading of core design. The predicted and measured values are compared in Table 4 which shows that the mean difference is 19.3 ppm. The agreement observed is well within the acceptance criterion of 50 ppm. Table 5 shows the calculated ITC against the measured ones. The agreement is again excellent with a mean difference of only $^{+}1.09$ C.

Usually measurements of control bank worths are made by one of the two methods, boron dilution and rod swap techniques. In the dilution method, the control bank worth is obtained using reactivities

Table 4. End Point Critical Boron Concentrations at Hot Zero Power Conditions

Bank	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5
ARO	5.8	17.5	21.3	30.0	29.0
D	5.8	–	13.6	25.6	31.9
C	−5.8 (D+C)	–	12.5	39.1	17.5
B	−17.3 (D+C+B)	–	−7.3	20.2	22.3
A	−15.9 (D+C+B+A)	–	8.7	24.1	33.5

(Δppm = meas. −calc.)

Table 6. Deviation between Calculated and Measured Bank Worths

Bank	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5
D	7.6	0.1	13.9	4.9	9.9
C	7.0 (D+C)	–	4.0	7.3	5.2
B	7.8 (D+C+B)	–	10.0	7.8	8.8
A	−2.4 (D+C+B+A)	–	–	8.8	8.8

(Δpcm = (calc. −meas.)/meas.x 100.)

Table 5. Comparison of Calculated and Measured Isothermal Temperature Coefficients

BANK	Cycle 1		Cycle 2		Cycle 3		Cycle 4		Cycle 5	
	meas.	calc.	meas.	calc.	meas.	calc.	meas.	calc.	meas.	calc.
ARO	6.75	6.9	−1.73	−2.5	0.57	−1.6	1.13	−0.1	−1.07	0.3
D in	7.25	7.9	−1.72	−2.7	−0.78	−2.7	1.26	−0.9	–	–

(pcm/°C)

caused by the difference between the initial and final critical boron concentrations of the moving control rod bank. The rod swap method is to get the bank worth by exchanging the test banks with the reference bank while keeping the boron concentration. The MEDIUM3 calculations simulated the control rod bank configurations according to the method employed. The deviations between predictions and measurements are given in Table 6. The overall agreement in individual bank worth is satisfactory except cycle 3 as the typical acceptance criterion is less than 10%.

5.2. Core Follow Prediction

Core follow prediction encompasses two main measurements as part of reactor operation: critical boron concentration and reaction rate distribution of incore detectors measured during core cycle irradiation. As a whole the predictions in critical boron concentrations agree with measurements within an acceptance criteria of 50 ppm. The average absolute difference is, over 60 data points covering the five core cycles, 34 ppm with a standard deviation of 15 ppm. Fig. 5 shows the letdown of critical boron concentrations versus core cycle burnup for YGN−1 cycle 5.

The movable fission chambers measure reaction rates at selected core positions. Cycle-specific precalculated values of power-to-reaction rate constants are then used to convert the detector readings to relative core power distributions. Thus re-

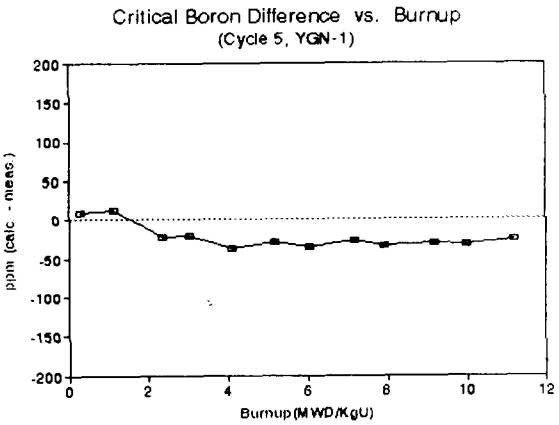

Fig. 5. Deviation between Calculated and Measured Critical Boron Letdown at Cycle 5

action rate measurements enable a more direct and accurate verification of calculated 3−D flux solutions. The comparisons of reaction rate distributions were only performed for a subset of total cycles analyzed. Table 7 shows deviations between calculations and measurements of axially integrated reaction rate distributions. The agreement is excellent with an average difference of only 1.4%. An illustration of radial reaction rate distributions is given in Fig. 6.

6. Conclusions

An attempt has been made here to develop and

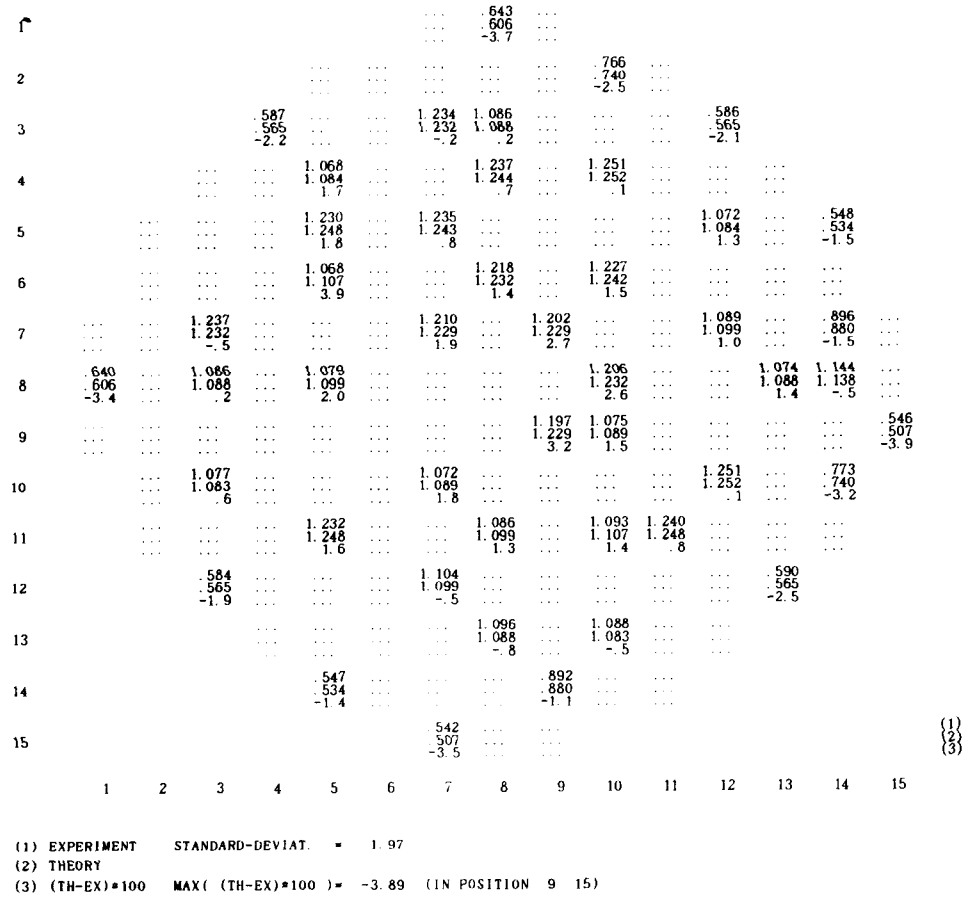

Fig. 6. Comparison of Calculated and Measured Detector Reaction Rates at 11.400 MWD/KgU of Cycle 1

Table 7. Deviations between Calculated and Measured Detector Reaction Rates

Burnup	Cycle 1		Cycle 2		Cycle 3		Cycle 4	
(MWD/KgU)	1.712	11.400	0.868	10.935	1.550	11.000	3.460	10.976
standard deviation	1.87	1.97	2.08	1.18	1.17	0.96	1.18	1.04
maximum difference	6.14	−3.89	4.33	3.40	−3.51	−2.78	−2.81	−1.96

(Difference = (calc. −meas.)x100)

validate the nuclide chain model best suitable for CASMO3/MEDIUM3 procedure as an independent option of SAV90 for the PWR design and fuel management calculations. To this end four candidates of nuclide chain models were constructed in such a way that they became consistent with CASMO3 chains.

The proposed models have been evaluated for various fuel types presently loaded in PWRs, based on the reproduction accuracy of MEDIUM 0D calculations in assembly k_∞, reactivity coefficients, and control rod worth. The results showed that for practical design application the 22 nuclide model was recommended in terms of accuracy of reproduction and computing efficiency. Increasing the complexity of nuclide chains beyond this did not improve the accuracy any further within the desired criteria of 0.1% Δk. The 25 nuclide chain model, however, may be a better choice in case the effect of higher actinides such as Am, Cm becomes important.

The 22 nuclide model has been further validated against the measured data for cycles 1 to 5 of YGN−1. The measurements for YGN−1 provide broad data basis for the verification of CASMO3/MEDIUM3/PINPOW2, since its operating history encompasses wide range of fuel types and fuel management schemes. Core follow calculations included simulation of startup physics tests and core cycle operations. For all the physics parameters compared, the deviations between predictions and measurements turned out to be well within the typical design criteria. This close agreements clearly demonstrated successful adaptation of the SAV standard nuclide chain for CASMO3/MEDIUM3 procedure.

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References

1. K. Koebke et al., "The Core Design Procedure, SAV90, for Pressurized Water Reactors," Kerntechnik, 57(1), 37 (1992)
2. K. Koebke et al., "Topical Report on Nuclear Design Procedure SAV79A," KWU Report, Sept. (1987)
3. M. Edenius, B-H Forssen, and H Haeggblom, "CASMO-3: A Fuel Assembly Burnup Program Methodology," STUDSVIK/NFA−89/2 (1989)
4. Y.J. Kim, "Verification of CASMO3/MEDIUM3 by Comparison with the Standard SAV90 Procedure," KWU BT21/92/E515 (1992)
5. Koebke et al., "The Equivalent Reflector Model," KWU R171/82/307 (1982)
6. K. Koebke, H. Haase, L. Hetzelt, H. -J. Winter, "Application and Verification of the Simplified Equivalence Theory for Burnup States," Nucl. Sci. Eng. 92, p. 56 (1986)