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SIMULATION OF THE POWER FEEDBACK EFFECTS WITH THE REACTIVITY COEFFICIENT METHOD

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ABSTRACT

Reactivity coefficient method offers a way to correct macroscopic diffusion cross-sections when operating conditions change. The method is based on reactivity coefficients. Coefficients are not dependent on a reactor operating history. They can be calculated in advance and stored in a library. The library needs to be updated only when drastic changes in the fuel design (different dimensions, new type of burnable poisons, etc.) are introduced. Two ways of the implementation are investigated in the paper. In the first one the nodal diffusion solver is extended with the power feedback option. This allows calculation of a reactor core at arbitrary conditions with only one basic set of macroscopic cross-sections. The second one is a possibility to generate a cycle specific macroscopic cross-section provider. The cross-section library is generated with the standard calculational procedures at three specific operating conditions (hot full power, hot zero power and cold zero power) and at several cycle burnup steps. Cross-sections at any other operating conditions are obtained by modification of the cross-sections with the reactivity coefficient method and suitable interpolation. The method is tested on several realistic cases, which usually arise in the process of nuclear core design. The method has proved to be a powerful tool in the simulation of the power feedback effects.

1. INTRODUCTION

The standard procedure in the analysis of reload cores is to generate a library of neutron macroscopic cross-sections. Since the library is prepared before the actual core characteristics are known, some average fuel assembly operating conditions must be assumed. A more accurate approach adopted in the CORD-2[1] package is to rely on the isotopic composition library. At each burnup step only the cell isotopic composition is retrieved from a precomputed library, based on the actual operating conditions. Everything else (cell cross-sections determination, nodal cross-section homogenisation, global diffusion calculation) is performed on the case by case basis. The

advantage of the process is that the influence of the actual assembly power history on the isotopic composition and consequently on cross-sections is taken into account. However, after the core cycle depletion is done, only isotopic composition and nominal Hot Full Power (HFP) macroscopic cross-sections are available. Since the analysis of reload cores normally requires examination of several different cases, a method to correct cross-sections for arbitrary operating conditions is needed if one wants to avoid excessive cross-sections recalculation. The Reactivity Coefficient Method (RCM) proved to be very effective in that respect.

The method offers several possibilities of application. Nodal diffusion codes LEM[2] and GNOMER[3] have been extended with the power feedback option which enables dynamic reactor calculations. Codes are part of the CORD-2 system used for the design calculation of reload cores. GNOMER is also the diffusion solver of the LOADF[4] core surveillance package running on the Process Information System of the Krško power plant simulating in the real time reactor operations.

RCM has been also incorporated in the stand alone cycle-dependent macroscopic cross-section library, which provides cross-sections at arbitrary core conditions. The library is used for scoping studies in the phase of preliminary loading pattern evaluation and serves on the NPP Krško full scope simulator as a main provider of cross-sections.

2. REACTIVITY COEFFICIENT METHOD

2.1 REACTIVITY MODEL

The method is based on the reactivity coefficients[5], which describe reactivity changes due to the following effects:

- Moderator temperature (T_M),
- Fuel temperature (T_F),
- Boron concentration (C_B),
- Xenon (Xe).

Coefficients represent fuel behaviour and are a function of the operating conditions and fuel characteristics. They are normally stored in a library and are, beside operating conditions, also tabulated as a function of burnup (Bu), enrichment (Enr) and number of burnable absorbers (BA) present in a node. The library needs to be updated only when drastic changes in the fuel design (different dimensions, new type of burnable poisons, etc.) are introduced.

Reactivity changes on a node are approximated by:

$$\Delta\rho_B = \int_{C_B^1}^{C_B^2} \frac{\partial\rho(C_B, T_M^1, Bu, Enr, BA)}{\partial C_B} dC_B \quad (1)$$

$$\Delta\rho_M = \int_{T_M^1}^{T_M^2} \frac{\partial\rho(T_M, C_B^2, Bu, Enr, BA)}{\partial T_M} dT_M \quad (2)$$

$$\Delta\rho_F = \int_{T_F^1}^{T_F^2} \frac{\partial\rho(T_F, T_M^2, Bu, Enr, BA)}{\partial T_F} dT_F \quad (3)$$

$$\Delta\rho_{Xe} = \int_0^{Xe^2} \frac{\partial\rho(Xe, C_B^2, Bu, Enr, BA)}{\partial Xe} dXe - \int_0^{Xe^1} \frac{\partial\rho(Xe, C_B^1, Bu, Enr, BA)}{\partial Xe} dXe \quad (4)$$

2.2 SPECTRAL CORRECTIONS

Since the method is intended to cover broad range of operating conditions, a change in the infinite neutron spectrum and in the leakage have to be considered. Those effects usually represent smaller corrections than reactivity changes and are for the moment considered to be dependent only on operating conditions and geometry of the problem (fuel to moderator ratio). In the two energy group approximation, selected cross-sections are corrected with the following factors:

Table I: Spectral Corrections

Spectral factor	Affected cross-sections
C_{spect}	Σ_{tr1}, Σ_{a1}, Σ_{12}, $\nu\Sigma_{f1}$, Σ_{f1}
C_{dif}	Σ_{tr1}, Σ_{tr2}
C_{12}	Σ_{12}
C_{frac}	Σ_{a1}, Σ_{a2}

After spectral corrections (C_{spect}, C_{dif}, C_{12}), the absorption cross-sections are modified to reproduce correctly the node K_∞ in such manner that the fast to thermal absorption increment ratio equals to C_{frac}. All spectral correction factors C are fitting parameters generated to match the observed differences in cross-sections when core conditions change. All cross-section modifications are done for each node individually.

3. RESULTS

The method is tested on several realistic cases of the Krško NPP. Krško is a 2-loop Westinghouse PWR. The reactor core consists of 121 fuel assemblies with 16X16 rod array. Krško is going to replace the existing steam generators and uprate reactor thermal power by 6.3%. Two typical 1 year reload cycles – uprated and non-uprated, with approximately one-fourth fresh fuel loading, are selected. In the non-uprated cycle only fuel with enrichment 4.3 % is used, while a fresh fuel batch in the uprated cycle consists of 4.6% fuel.

3.1 NODAL DIFFUSION CODE

Testing was performed with the LEM nodal diffusion code. Starting with a Hot Zero Power (HZP) no Xe cross-sections, Hot Full Power (HFP) core conditions are simulated using the reactivity coefficient method. Results of calculations (marked as LEM) are compared with results based on the standard sequence of calculation with the CORD-2 package (marked as CORD-2) in which the cross sections are generated explicitly for the HFP case.

Comparison of the critical boron concentrations for the Beginning, Middle and End Of Cycle (BOC, MOC, EOC) conditions is shown in Table II. Maximal difference is 15 ppm and lies well within standard 50 ppm review criteria.

Maximum and average difference in assembly power for the BOC, MOC and EOC can be found in Table III. Maximum errors εP_{MAX} are less then 1.5% and average difference εP_{AVE} is less than 0.8 %. Assemblywise HFP power and error distributions are presented in Figures 1 (non-uprated cycle) and 2 (uprated cycle).

Table II: Comparison of the HFP Critical Boron Concentrations at the BOC, MOC and EOC for the Non-uprated and Uprated cycle.

	Non-uprated			Uprated		
	CORD-2 [ppm]	LEM [ppm]	DIFF. [ppm]	CORD-2 [ppm]	LEM [ppm]	DIFF. [ppm]
BOC	1339	1337	-2	1358	1343	-15
MOC	653	657	+4	848	840	-8
EOC	49	55	+6	60	53	-7

Table III: Maximal and Average Difference in Assembly Power for the BOC, MOC and EOC (Non-uprated, Uprated cycle).

	Non-uprated		Uprated	
	εP_{MAX} [%]	εP_{AVE} [%]	εP_{MAX} [%]	εP_{AVE} [%]
BOC	0.47	0.27	1.26	0.62
MOC	0.68	0.36	1.30	0.68
EOC	0.96	0.50	1.48	0.78

3.2 CYCLE-DEPENDENT MACROSCOPIC CROSS-SECTION PROVIDER

The macroscopic cross-section library for the non-uprated cycle is generated with the standard CORD-2 calculational procedure at three specific operating conditions (HFP, HZP and CZP - Cold Zero Power) and at several cycle burnup steps. Nodal cross-sections at any other operating conditions are obtained by modification of the cross-sections with the reactivity coefficient method and suitable interpolation.

3.2.1 ZERO POWER CONDITIONS

Comparison of the critical boron concentrations for the All Rods Out (ARO) and All Rods In (ARI) configuration at the BOC, MOC and EOC conditions are shown in Tables IV and V. Temperature was varied over the entire CZP-HZP range. Maximal difference for the ARO condition is 9 ppm and 22 ppm for the ARI case. Observed differences are within the accuracy of the CORD-2 package and lie well within standard 50 ppm review criteria.

Table IV: Comparison of the ARO Critical Boron Concentrations at BOC, MOC and EOC.

T [C]	BOC			MOC			EOC		
	CORD-2 [ppm]	RCM [ppm]	DIFF. [ppm]	CORD-2 [ppm]	RCM [ppm]	DIFF. [ppm]	CORD-2 [ppm]	RCM [ppm]	DIFF. [ppm]
21	2110	2110	0	1650	1650	0	1234	1233	-1
60	2117	2123	6	1650	1656	6	1226	1232	6
93	2127	2135	8	1649	1658	9	1216	1225	9
149	2150	2157	7	1647	1653	6	1192	1196	4
177	2163	2168	5	1643	1645	2	1174	1172	-2
204	2170	2173	3	1631	1630	-1	1144	1140	-4
260	2169	2172	3	1569	1570	1	1034	1033	-1
292	2137	2137	0	1482	1482	0	905	904	-1

Table V: Comparison of the ARI Critical Boron Concentrations at BOC, MOC and EOC.

T [C]	BOC			MOC			EOC		
	CORD-2 [ppm]	RCM [ppm]	DIFF. [ppm]	CORD-2 [ppm]	RCM [ppm]	DIFF. [ppm]	CORD-2 [ppm]	RCM [ppm]	DIFF. [ppm]
21	1370	1365	-5	912	912	0	528	533	5
60	1349	1354	5	883	893	10	492	508	16
93	1327	1334	7	850	863	13	452	471	19
149	1275	1278	3	774	780	6	356	369	13
177	1240	1237	-3	724	723	-1	293	299	6
204	1192	1185	-7	656	652	-4	210	215	5
260	1031	1028	-3	438	442	4	-49	-32	17
292	858	853	-5	214	221	7	-309	-287	22

3.2.2 AT POWER CONDITIONS

Comparison of the total power defect is presented on Table VI. Observed differences are very small. Results of the HFP control rod worth calculations are shown in Table VII. Even for such extreme case as total insertion of control rods at HFP, the agreement is excellent.

Power distribution errors for the core at 75% nominal power are presented on Figure 3. Agreement is again excellent; differences are smaller than 0.25%.

Table VI: Comparison of the Total Power Defect at BOC, MOC and EOC.

P [%]	BOC			MOC			EOC		
	CORD-2 [pcm]	RCM [pcm]	DIFF. [pcm]	CORD-2 [pcm]	RCM [pcm]	DIFF. [pcm]	CORD-2 [pcm]	RCM [pcm]	DIFF. [pcm]
100	-1953	-1963	-10	-2402	-2415	-13	-2971	-3015	-44
75	-1513	-1522	-9	-1839	-1855	-16	-2264	-2308	-44
50	-1045	-1053	-8	-1273	-1290	-17	-1562	-1605	-43
25	-544	-550	-6	-667	-685	-19	-823	-850	-28
0	0	0	0	0	0	0	0	0	0

Table VII: Comparison of the HFP Control Rod Worths at BOC, MOC and EOC.

Control bank	BOC			MOC			EOC		
	CORD-2 [pcm]	RCM [pcm]	DIFF. [pcm]	CORD-2 [pcm]	RCM [pcm]	DIFF. [pcm]	CORD-2 [pcm]	RCM [pcm]	DIFF. [pcm]
D	753	751	-2	797	796	-1	817	816	-1
C	1238	1239	1	1224	1225	1	1266	1266	0
B	944	950	6	1030	1034	4	1078	1083	5
A	1566	1569	3	1826	1828	2	1860	1859	-1
S	3855	3872	17	3924	3933	9	4012	4002	-10

4. CONCLUSIONS

Reactivity coefficient method presented in the paper has proved to be a powerful tool in the simulation of the power feedback effects. The method can be used to extend the capability of the static diffusion codes to dynamic reactor calculation or can be a basic tool in the generation of the cycle dependent macroscopic cross-section library. It has been tested on several realistic cases of the NPP Krško. Results have shown that the method fulfills even the strongest requirements of the nuclear design calculations.

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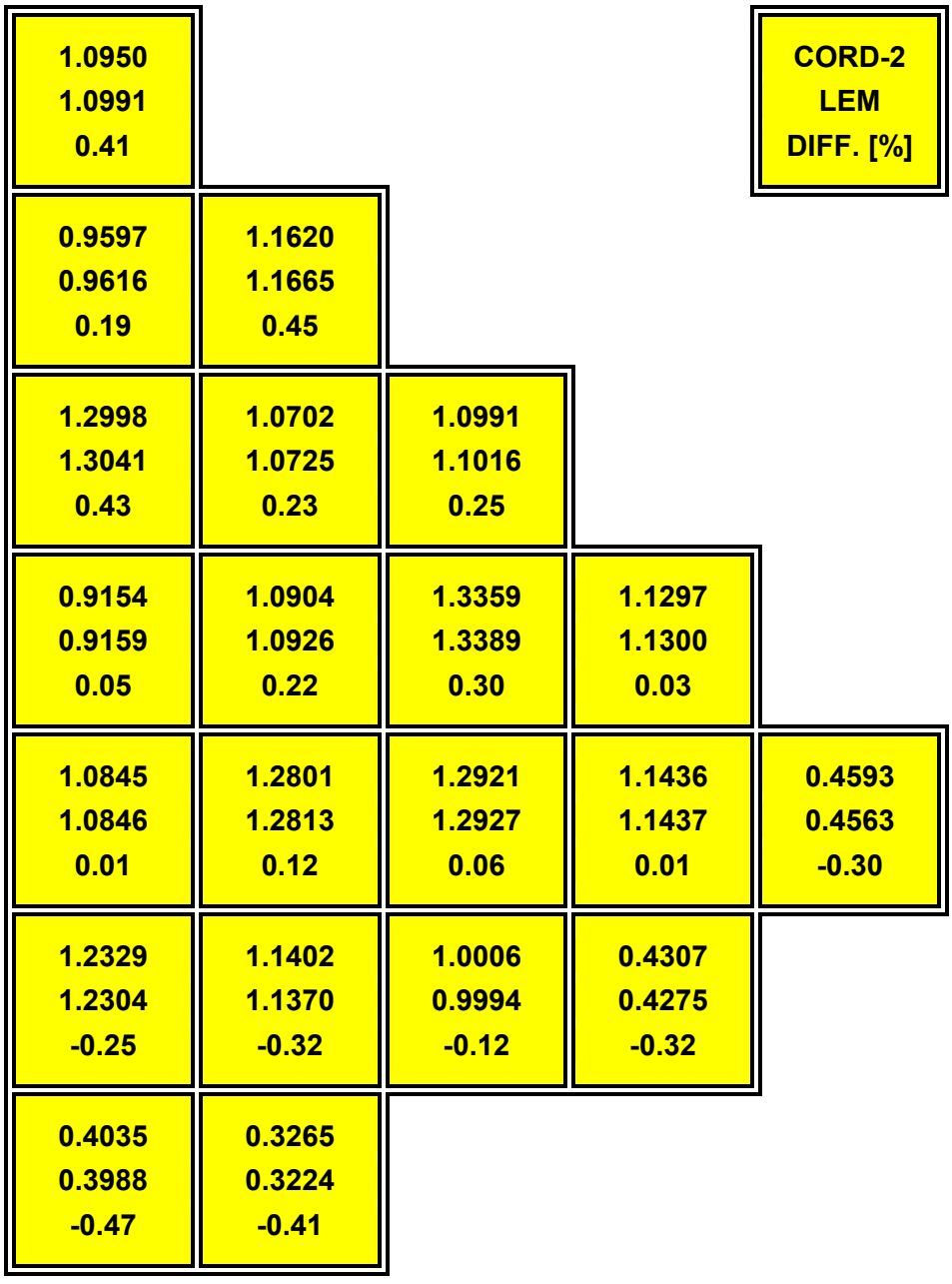

Figure 1: Comparison of the Assemblywise Power at the BOC, Non-uprated Cycle.

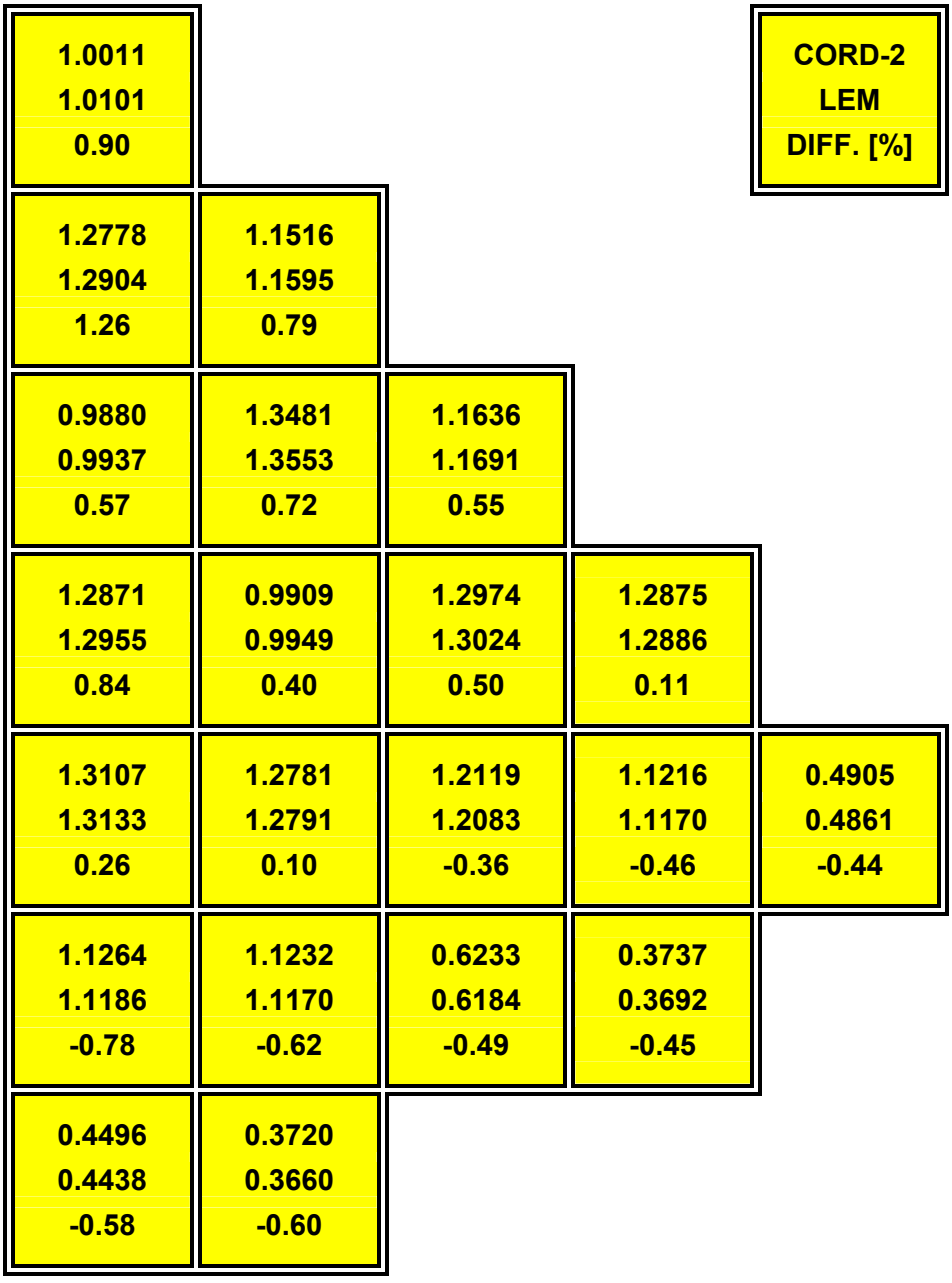

Figure 2: Comparison of the Assemblywise Power at the BOC, Uprated Cycle.

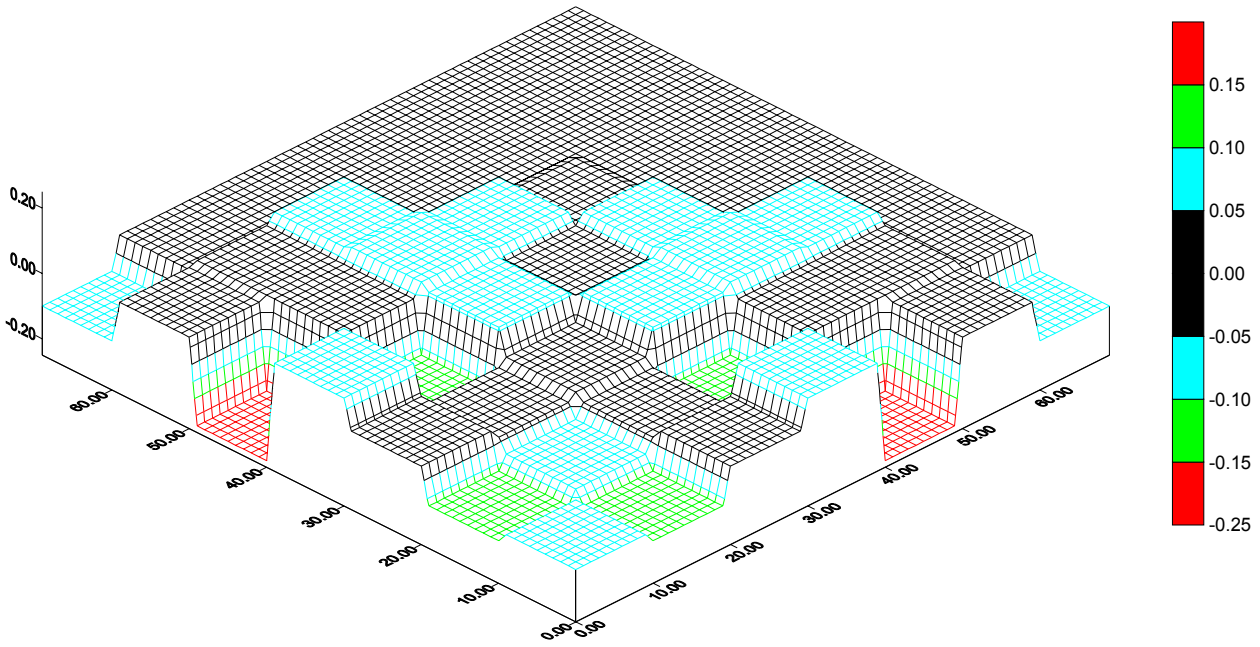

Figure 3: Power Distribution Differences (in % with maximum 0.23%) at the 75 % Core Power Level, BOC.