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PARAMETERIZATION OF TWO-GROUP NODAL CROSS SECTION DATA FOR POLCA-T BWR TRANSIENT APPLICATIONS

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ABSTRACT

During the last decade, the use of coupled three-dimensional nodal neutron kinetics and thermal hydraulics simulation tools has gained increased popularity in performing comprehensive nuclear reactor design and safety analyses for BWRs. One of the advantages of using these coupled codes is the capability of providing full consistency between static core analyses and corresponding transient safety analyses. Therefore, the quality and accuracy of the underlying nodal method including the nodal cross section representation model play an important role in the successful application of these multi-physics tools. In this paper, the cross section parameterization required for BWR transient applications is considered with the objective to find a representation that provides complete transparency to users with regard to both steady-state and transient core analyses. It is demonstrated by means of simple 2D unit assembly test cases that if coolant boiling occurs below hot core operation conditions one needs to include both the coolant density and the moderator temperature as explicit state parameters in the cross section tabulation.

Key Words: Coupled Nodal Codes, Nodal Cross Section Representation Models

1. INTRODUCTION

During the last decade, the use of coupled three-dimensional (3D) nodal neutron kinetics and thermal hydraulics simulation tools has gained increased popularity in performing comprehensive nuclear reactor design and safety analyses for boiling water reactors (BWRs). At Westinghouse the POLCA-T code [1] serves this purpose as it encapsulates the advanced nodal core simulator POLCA7 [2] and the most advanced features of available in-house thermal hydraulics system transient codes into a single software product.

One important objective and advantage of these coupled codes is the capability of providing full consistency between static core analyses and corresponding transient safety analyses. Consequently, the quality and accuracy of the underlying nodal method including the nodal cross section representation model play an important role in the successful application of these multi-physics tools for analyzing core dynamics of a BWR.

Recently, the importance of having an appropriate cross section parameterization for light water reactor (LWR) transient applications [3,4,5] has been recognized. In this paper, the cross section parameterization required for BWR transient applications is discussed. In particular, the treatment of the coolant density versus moderator temperature is discussed with regard to the differences between the models that are employed in static and transient calculations, respectively. In static nodal core analysis applications the cross section dependence on moderator temperature is obtained implicitly via the dependence on the coolant density assuming saturated conditions with no boiling allowed below hot coolant conditions. However, during certain BWR transients, such as a nuclear heating flashing event (NHFE) or an anticipated transient without control rod insertion (ATWC), coolant boiling at moderator temperatures below hot coolant conditions will occur. At these so-called “transient” coolant conditions, it is necessary to decouple the nodal cross section dependence on the moderator temperature from the coolant density as will be demonstrated by means of numerical results in this paper.

Approaches varying from the very simple to the rather complicated have been used to cater for the moderator temperature and coolant density dependence of the nodal cross sections in transient applications. For instance, Demazière [4] applies a moderator density dependent linear correction term with regard to the moderator temperature whereas Watson et al. [3] uses a model that selects between the pressure and the coolant void to represent coolant density in various coolant condition regimes (single-phase versus two-phase coolant). The first approach, mainly applied for pressurized water reactor (PWR) transients,[5] is considered to simplistic to handle all BWR transient scenarios as linear correction terms in combination with multi-dimensional linear interpolation are employed for cross sections with respect to instantaneous state parameters. The second approach is considered unnecessarily complicated in that separate cross section libraries for the different coolant density regimes are required. In this paper we present a unified cross section representation model covering the complete core operation and transient moderator temperature and coolant density ranges within a single cross section tabulation. This is accomplished using moderator temperature and coolant density explicitly as state parameters in both the single-phase and two-phase regimes. The major advantage of this approach is that a single cross section library can be generated and used in both steady-state and transient core analyses providing complete transparency to users of the POLCA7 and POLCA-T codes.

The layout of this paper is as follows. In Section 2 a short overview of the cross section model employed by POLCA7 and POLCA-T is given. In Section 3, numerical results are provided for simple two-dimensional (2D) test problems. Finally, in Section 4 some conclusions are given.

2. CROSS SECTION MODEL OF POLCA7 AND POLCA-T

The static cross section model of POLCA7 is based on a Taylor series expansion around some hot-full-power (HFP) nominal condition denoted the “base condition” where each correction term in this expansion is parameterized in terms of the fuel exposure E, the momentaneous coolant density ρ and the coolant density history ρ_h:

$$\Sigma\left(E,\rho_h,\rho,SG,CR,T_f,N_i,C_B,\phi\right)=\Sigma^{base}\left(E,\rho_h,\rho\right)+\frac{\partial\Sigma\left(E,\rho_h,\rho\right)}{\partial CR}\Delta CR+\frac{\partial\Sigma\left(E,\rho_h,\rho\right)}{\partial SG}\Delta SG$$
$$+\frac{\partial\Sigma\left(E,\rho_h,\rho\right)}{\partial\sqrt{T_f}}\left(\sqrt{T_f}-\sqrt{T_f^{ref}}\right)$$
$$+\frac{\partial\Sigma\left(E,\rho_h,\rho\right)}{\partial C_B}\left(C_B-C_B^{ref}\right)+\frac{\partial\Sigma\left(E,\rho_h,\rho\right)}{\partial N_{Xe}}\left(N_{Xe}-N_{Xe}^{ref}\right)$$
$$+\sum_i\left[\sigma_i^{base}\left(E,\rho_h,\rho\right)+\frac{\partial\sigma_i\left(E,\rho_h,\rho\right)}{\partial CR}\Delta CR+\ldots\right]\left(N_i-N_i^{ref}\right)$$
$$+\text{ higher order and cross terms}$$
$$+\text{ intra-nodal correction terms}$$

The additive deviation terms are computed by varying the various state parameters individually or in unison from their nominal values, i.e. by perturbing the base state. The moderator temperature and coolant density combinations utilized in static cross section generation are specified in Tab. I and are illustrated in Fig. 1 (for the purpose of comparisons this set of nodal cross section data is referred to as “static cross section tables” in this paper).

Table I. Static coolant conditions.

	Saturated values			**Sub-cooled values**	
Temp [C]	**Press [bar]**	**Void [%]**	**Dens ρ [g/cm^3]**	**Press [bar]**	**Dens ρ [g/cm^3]**
20				1.01345	0.998232
60				1.01345	0.983200
100	1.01345	0.0	0.958388		
150	4.8	0.0	0.917060		
210	19.2	0.0	0.852829		
286	70	0.0	0.739911		
286	70	30.0	0.528898		
286	70	60.0	0.317885		
286	70	80.0	0.177209		

As an example of the impact on nodal cross sections of moderator temperature variations at fixed coolant density, the infinite multiplication factor as computed by a lattice code for a fresh assembly is shown in Fig. 2. Clearly, the static cross section model, which corresponds to the right-most points (at T_m = 286 C) in this figure, is incapable of adequately capturing the physics of the transient states illustrated in this figure. The transient cross section model for POLCA-T applications was therefore constructed to contain both the moderator temperature and coolant density as explicit state parameters for all microscopic and macroscopic nodal cross sections as well as corresponding coefficients (i.e. the three-dimensional tables of the static model was extended to four-dimensional tables):

$$\Sigma\left(E,\rho_h,\rho,T_m,SG,CR,T_f,N_i,C_B,\phi\right)=\Sigma^{base}\left(E,\rho_h,\rho,T_m\right)+\frac{\partial\Sigma\left(E,\rho_h,\rho,T_m\right)}{\partial CR}\Delta CR+\ldots$$

In this cross section tabulation the coolant density represents macroscopic effects of neutron moderation whereas the moderator temperature represents neutron thermalization effects. [6]

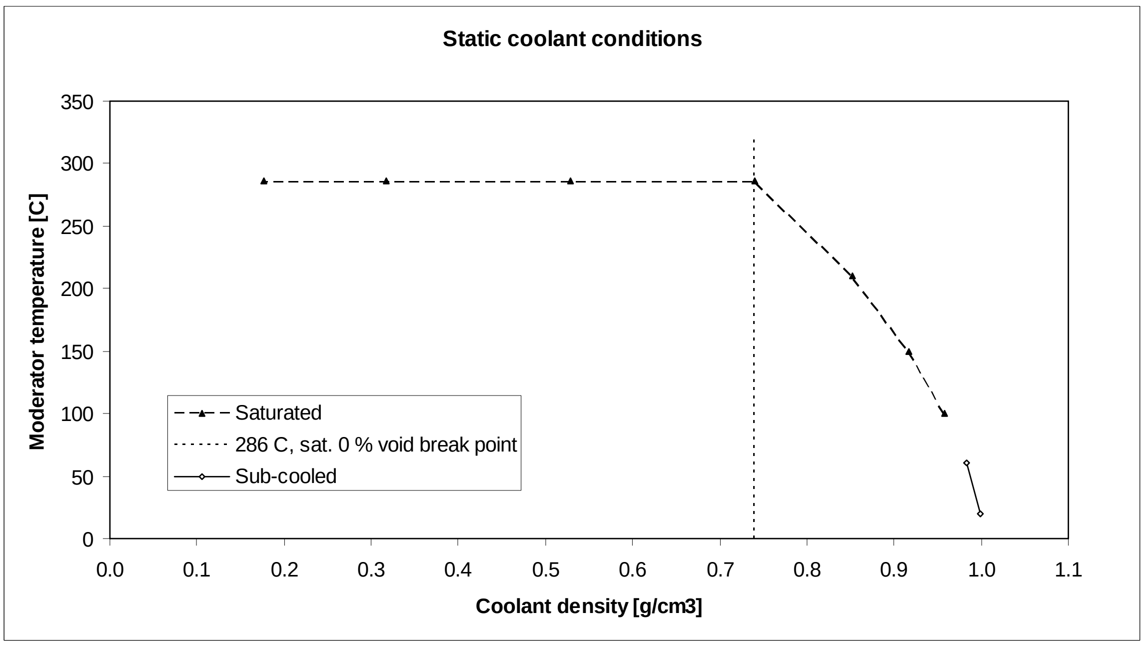

Figure 1. Coolant conditions applied in cross section data generation for static applications.

The moderator temperature and coolant density combinations utilized in transient cross section generation are given in Tab. II and III as well as illustrated in Fig. 3 (for the purpose of comparisons this set of nodal cross section data is referred to as “transient cross section tables” in this paper). Note that the static cross section model state points are now a sub-set of the transient state points (with the exception of the state points at moderator temperatures below 100 C which are generated at a pressure of 1 bar for the static cross section tables).

The following rules have been followed when choosing appropriate transient coolant conditions:

- In order to obtain realistic irradiation conditions and spectrum histories representative of nominal core operation, all **lattice depletion calculations** must be performed at HFP (voided and unvoided) coolant conditions at saturated pressure. These so-called HFP coolant density histories should constitute a subset of all considered momentaneous coolant densities in the lattice calculations.

- For each coolant density value there must be a set of momentaneous moderator temperatures covering the whole range of transient conditions (i.e. several moderator temperatures for each coolant density).
- For each moderator temperature there should be a coolant density entry corresponding to saturated 0 % void condition in order to avoid extrapolation from the saturated to the sub-cooled regime.

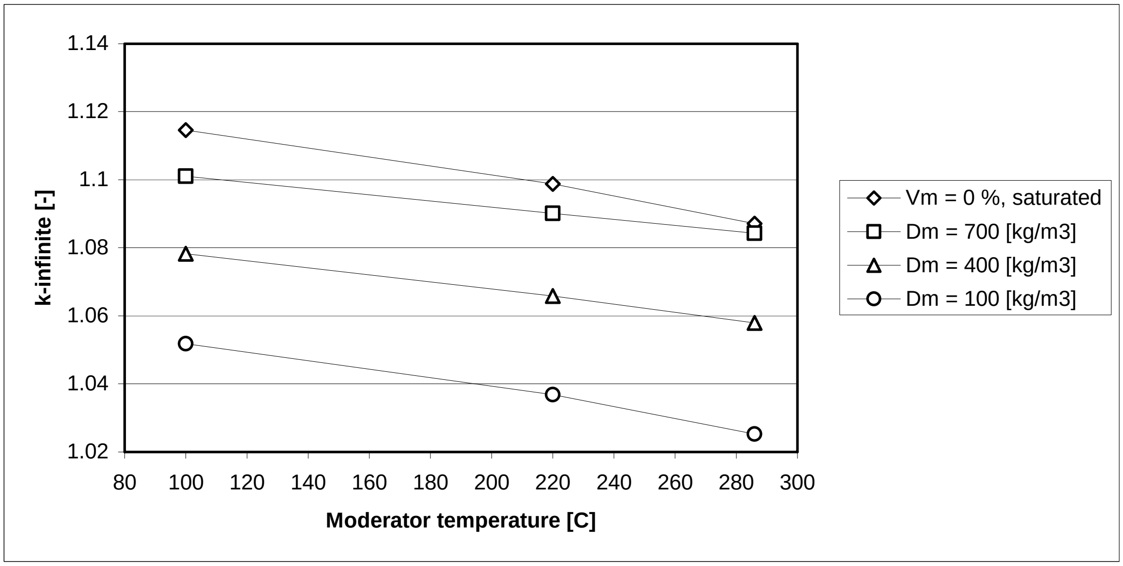

Figure 2. Dependence of k-infinity on moderator temperature for some typical fixed coolant densities.

Table II. Voided transient coolant conditions.

Saturated values				**Coolant density entries ρ [g/cm^3]**				
Temp [C]	**Liq. Dens [g/cm^3]**	**St. Dens [g/cm^3]**	**Press [bar]**	**Void corresponding to the coolant density entries of 0.10, 0.25, 0.40, 0.55 and 0.70 g/cm^3 [%]**				
				ρ = 0.10	ρ = 0.25	ρ =0.40	ρ =0.55	ρ =0.70
20	0.998187	1.73E-05	0.02339	89.98340	74.95589	59.92839	44.90088	29.87338
60	0.983162	0.00013	0.19935	89.84064	74.58173	59.32281	44.06389	28.80497
100	0.958388	0.000598	1.01345	89.62170	73.96065	58.29960	42.63856	26.97751
140	0.926183	0.001965	3.612	89.39262	73.16269	56.93276	40.70283	24.47290
180	0.887059	0.005154	10.02	89.24536	72.23672	55.22808	38.21944	21.21080
220	0.840331	0.011608	23.18	89.33390	71.23377	53.13364	35.03352	16.93339
260	0.783814	0.023703	46.9	89.96240	70.22844	50.49447	30.76051	11.02655
286	0.739911	0.036534	70	90.97692	69.65124	48.32556	26.99988	5.674195
300	0.712397	0.046158	85.845	91.91856	69.40410	46.88965	24.37520	1.860745

Table III. Saturated 0 % void and sub-cooled transient coolant conditions.

	Saturated 0% void values		Sub-cooled values	
Temp [C]	**Press [bar]**	**Dens ρ [g/cm³]**	**Press [bar]**	**Dens ρ [g/cm³]**
20	0.02339	0.998187	105	1.00294
60	0.19935	0.983162	105	0.987698
100	1.01345	0.958388	105	0.963208
140	3.612	0.926183	105	0.931589
180	10.02	0.887059	105	0.893276
220	23.18	0.840331	105	0.847389
260	46.9	0.783814	105	0.791131
286	70	0.739911	105	0.7458
300	85.845	0.712397	105	0.716672

As regards the interpolation in the multi-dimensional tables it is noted that interpolation in coolant density is performed prior to interpolation in moderator temperature. Consequently, it is possible to employ an irregular grid in coolant density for the cross section tables. A quadratic interpolation scheme with regard to fuel exposure, coolant density history and coolant density is used whereas linear interpolation is employed for the moderator temperature.

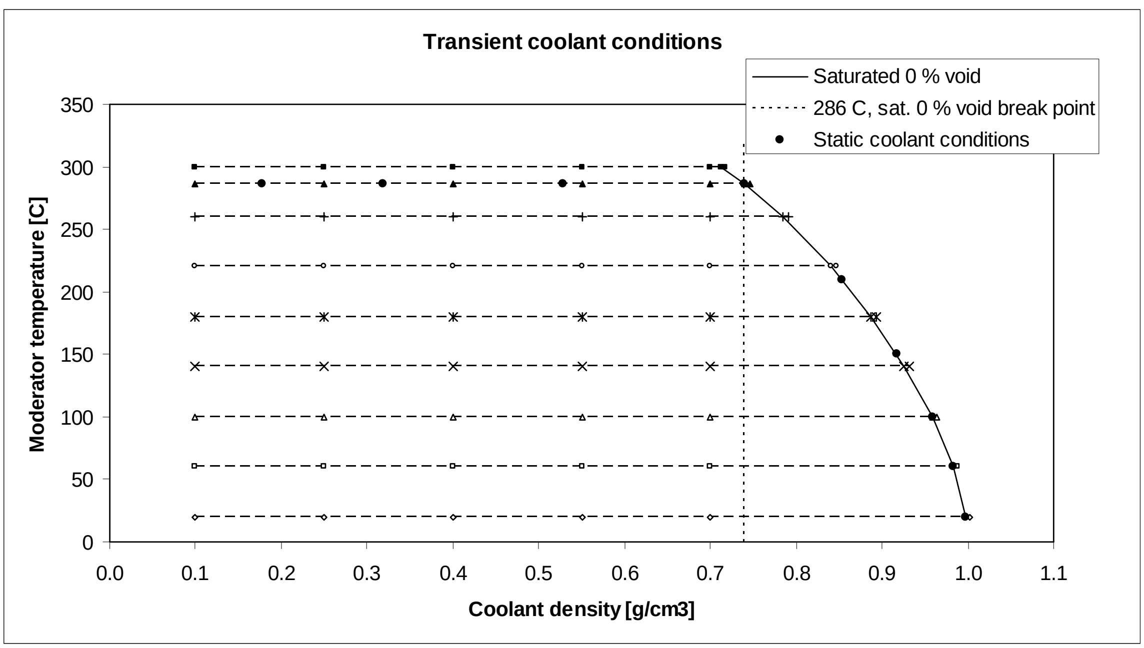

Figure 3. Coolant conditions applied in cross section data generation for transient applications.

3. NUMERICAL RESULTS

As a part of the verification and validation efforts initiated to qualify the transient cross section tables, a number of two-dimensional (2D) static test cases based on an infinite lattice have been evaluated with the nodal simulation tool CROSS* and compared to the corresponding results generated by the transport code PHOENIX4 [7]. In addition, an assessment of the accuracy of using the static cross section tables has also been made for these same test cases.

The unit assembly test cases evaluated in this paper represent some pre-selected scenarios typically obtained during normal BWR operation at rated power as well as during some transient events, both for fresh and irradiated assembly conditions. In particular, numerical results for fuel irradiation calculations as well as nuclear heating operation simulations are presented. Fuel depletion has been performed at typical HFP core conditions. A fuel assembly axial segment representative of a modern 10x10 BWR fuel design (Westinghouse SVEA-96 Optima2) has been modeled in these unit assembly† calculations for both rodded and unrodded conditions.

In this paper all numerical results are presented in terms of k-infinity as the reactivity is considered to be the most important parameter to track during transients. All errors in k-infinity are reported in units of [pcm] according to

$$\varepsilon_{k_{\text{inf}}} = \left(k_{\text{inf}}^{\text{CROSS}} - k_{\text{inf}}^{\text{PHX4}}\right)\times 10^{5} \ \ \text{[pcm]}$$

3.1. Fuel irradiation simulations

Fuel depletion calculations have been performed with the coolant void content set to both 40 % as well as 75 % employing two sets of depletion steps; the reference depletion steps built into POLCA7 as well as the depletion steps applied in lattice physics providing cross section data to the nodal simulator (in this paper referred to as “cell data generation”). Recall that the isotopic correction term accounts for any differences in number densities compared to those obtained for the reference depletion conditions. Except for xenon, these reference number densities for all tracked isotopes are determined internally by POLCA7 using the built-in reference depletion steps whereas the reference xenon number densities are provided by the lattice code according to:

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* The computer code CROSS is a nodal simulation tool which combines all the necessary routines and modules of POLCA7 to simulate (spatially homogeneous) unit assembly depletion and branch cases similar to those (spatially heterogeneous) cases employed to generate cross section tables for POLCA7 using PHOENIX4.

† Here the term unit assembly refers to a radial slice of an assembly with the size consistent with the core assembly pitch and infinite height.

$$\Delta\Sigma^{iso}\left(E_{cdstps},\rho_h,\rho,SG,CR,\ldots\right)=\frac{\partial\Sigma\left(E_{cdstps},\rho_h,\rho\right)}{\partial N_{Xe}}\left(N_{Xe}-N_{Xe}^{ref}\left(E_{cdstps},\rho_h\right)\right)$$
$$+\sum_i\sigma_i\left(E_{cdstps},\rho_h,\rho,SG,CR,\ldots\right)\left(N_i-N_i^{ref}\left(E_{refstps},\rho_h\right)\right)$$

In Tab. IV the depletion steps employed in the POLCA7 main depletion calculations of this study are summarized. It should be noted that comparisons and presentation of numerical results are performed at some selected fuel exposures common to both sets of depletion steps.

Table IV. Fuel depletion steps applied in fuel irradiation simulations.

Abbreviation	cdfstps	refstps
Application	cell data generation	base depletion internally in CROSS/POLCA7
Burn-up grid (E_1-E_2; ΔE) [MWd/tHM]	0-350; 350 350-500; 150 500-20000; 500 20000-24000; 1000 24000-70000, 2000	0-500; 100 500-1000; 250 1000-12000; 500 12000-20000;1000 20000-70000; 2000

In Tab. V the difference in k-infinity between CROSS and PHOENIX4 is shown when depleting the considered fuel assembly segment at a coolant void of 40 % using both transient cross section tables as well as static cross section tables. In Tab. VI results for the coolant void of 75 % are shown. The fuel exposure of the peak value in k-infinity is marked with grey in these tables.

Table V. Comparison of k-infinity between CROSS and PHOENIX4 using different sets of cross sections and depletion steps – depletion at a coolant void of 40 %.

Trans. XS			**Static XS**		
Burn-up	**cdfstps**	**refstps**	**Burn-up**	**cdfstps**	**refstps**
0	-10	-10	0	-2	-2
1000	-1	-9	1000	5	-3
5000	-12	-15	5000	-6	-9
11000	-16	-17	11000	-6	-7
20000	-45	-20	20000	-41	-16
28000	-81	-31	28000	-87	-36
40000	-81	-46	40000	-103	-68
50000	-89	-60	50000	-126	-97
60000	-90	-68	60000	-140	-118
70000	-84	-70	70000	-139	-125

Table VI. Comparison of k-infinity between CROSS and PHOENIX4 using different sets of cross sections and depletion steps – depletion at a coolant void of 75 %.

Trans. XS			Static XS		
Burn-up	**cdfstps**	**refstps**	**Burn-up**	**cdfstps**	**refstps**
0	-11	-11	0	-8	-8
1000	-1	-10	1000	0	-9
5000	0	-4	5000	-2	-6
12000	4	4	12000	-5	-5
20000	-34	-17	20000	-33	-16
28000	-63	-33	28000	-59	-30
40000	-77	-60	40000	-63	-47
50000	-107	-94	50000	-75	-62
60000	-135	-124	60000	-85	-75
70000	-156	-147	70000	-92	-84

As may be observed, the usage of both transient and static cross section tables gives similar results up to the exposure point at which the burnable absorber (BA) has fully depleted out (recognized in the tables as the grey-colored row corresponding to the peak k-infinity value). At higher burn-up values, these cross section setups differ more mainly due to the different accumulation of interpolation errors made in the involved correction terms of the cross section model. In this regard, it has been observed that the contribution from the actinide isotopic correction term dominates this accumulated error.

Application of reference depletion steps in the main depletion calculation generally gives smaller deviations compared to the cell data generation steps. This is mainly due to the fact that the reference number densities for actinides and fission products are generated employing these same depletion steps thereby minimizing the inconsistency between these two depletion calculations although the inconsistency in the xenon number density correction still remains. Apparently, the correction from heavy nuclides and fission products dominates the contribution from xenon in this particular case.

In summary, similar performance is recognized for fuel irradiation calculations using the transient cross section tables compared to the static cross section tables.

3.2. Nuclear heating simulations

In this section simulations at core conditions typical for nuclear heating operation are considered. In practice this means that isothermal conditions (i.e. $T_f = T_m$) are assumed in all cases going from cold up to hot zero power (HZP). Both non-voided and voided saturated coolant conditions have been evaluated where the latter represents a depressurization (i.e. flashing) event during core start-up. These nuclear heating calculations have been performed with CROSS and PHOENIX4 at two fuel exposures, namely, at 0 MWd/tHM with no xenon and at 12000 MWd/tHM with equilibrium xenon. Both rodded and unrodded configurations have been considered in this study.

In Tab. VII and VIII typical nuclear heating simulation results at unvoided rodded conditions are shown. As similar or more accurate results were obtained for corresponding unrodded cases these results are not shown in this paper. As may be seen, rather similar accuracy is obtained with both cross section table setups. The errors observed are mainly due to limitations in the POLCA7 cross section model, i.e. the lack of higher order terms as well as interpolation errors in case of the static cross section tabulation (see Tab. VII). It is also recognized that these errors increase with burn-up (see Tab. VIII) indicating that all spectrum effects induced by irradiation are not fully captured by the POLCA7 cross section model.

Table VII. Comparison of k-infinity between CROSS and PHOENIX4 using different sets of cross sections at 0 MWd/kg – unvoided rodded zero xenon conditions for T_m below 286 C.

Cold-HZP CROD Zero xenon (E = 0 MWd/kgHM)							
			K-infinite				
Mod. Temp. [C]	Press. [bar]	Dens. [g/cm3]	Trans. XS	Static XS	Reference	Err. Trans. XS [pcm]	Err. Static XS [pcm]
20	1.01345	0.998232	0.95933	0.95934	0.95945	-12	-11
60	1.01345	0.983199	0.95129	0.95128	0.95141	-12	-13
100	1.01345	0.958388	0.94186	0.94185	0.94198	-12	-13
140	3.612	0.926183	0.93163	0.9314	0.93175	-12	-35
180	10.02	0.887059	0.92034	0.91999	0.92046	-12	-47
220	23.18	0.840331	0.90712	0.90721	0.90721	-9	0
260	46.9	0.783814	0.89189	0.89195	0.89197	-8	-2
286	70	0.739911	0.88018	0.88015	0.88016	2	-1

Table VIII. Comparison of k-infinity between CROSS and PHOENIX4 using different sets of cross sections at 12 MWd/kg – unvoided rodded equilibrium xenon conditions for T_m below 286 C.

Cold-HZP CROD Eq. xenon (E = 12 MWd/kgHM)							
			K-infinite				
Mod. Temp. [C]	Press. [bar]	Dens. [g/cm3]	Trans. XS	Static XS	Reference	Err. Trans. XS [pcm]	Err. Static XS [pcm]
20	1.01345	0.998232	1.02572	1.02577	1.02455	117	122
60	1.01345	0.983199	1.02322	1.02328	1.02195	127	133
100	1.01345	0.958388	1.01933	1.01942	1.01804	129	138
140	3.612	0.926183	1.01438	1.01437	1.01315	123	122
180	10.02	0.887059	1.00789	1.00789	1.00674	115	115
220	23.18	0.840331	0.99935	0.99949	0.99846	89	103
260	46.9	0.783814	0.98756	0.98778	0.98726	30	52
286	70	0.739911	0.97712	0.97718	0.97735	-23	-17

Table IX and X report results for several voided unrodded states at each of a number of moderator temperatures below T_m = 286 C. The main qualitative observations with regard to these numerical results are the following:

- By using static cross section tables at moderator temperatures below T_m = 286 C for voided coolant conditions, the reactivity is strongly under-predicted at fresh core conditions (see Tab. IX). This is mainly due to the wrong thermal spectrum utilized for collapsing cross sections to two neutron energy groups. As static cross section generation presumes boiling at a saturated moderator temperature of 286 C, a harder thermal spectrum (approximate Maxwellian) will be frozen into these nodal two-group cross sections than would have been obtained with a lower moderator temperature.
- At fresh conditions in the absence of any fission products and parasitic absorption, the behaviour of the reactivity as a function of the moderator temperature is mainly governed by the actinide (in this case U-235 and U-238) cross section dependence on the neutron energy. By assuming a 1/v-dependence of the actinide fission cross sections in the thermal region, a lower reactivity will be built into the cross sections at T_m = 286 C compared to cross sections generated at lower moderator temperatures, as confirmed by the results presented in Tab. IX.
- As the moderation decreases with increased void content (i.e. the macroscopic effect of decreased coolant density), a further spectrum hardening occurs increasing the resonance absorption and consequently causing a further lowering of the reactivity. This effect is mainly seen for the nodal two-group cross sections generated at T_m = 286 C (i.e. static cross section tables) as in this case more neutrons will have energies close to the resonance region. Consequently, a further decrease in k-infinity with increased void content is obtained compared to the corresponding reference PHOENIX4 results using the static cross section tables.
- At irradiated conditions (see Tab. X), the nodal cross section response to moderator temperature changes will be different from the response at fresh conditions mainly due to the presence of fission products and fissile plutonium isotopes. As the absorption cross sections of most fission products, and especially xenon, show a 1/v-dependence on the neutron energy, a temperature increase will shift the thermal spectrum to higher energies reducing the parasitic absorption in the fuel thereby increasing the reactivity. Therefore an over-prediction of the reactivity is seen using the static cross section tables at temperatures below T_m = 286 C.

In summary, at non-voided coolant conditions both cross section table setups give the same level of accuracy whereas at voided coolant conditions below T_m = 286 C the transient cross section tables are greatly superior to the static tables.

Table IX. Comparison of k-infinity between CROSS and PHOENIX4 using different sets of cross sections at 0 MWd/kg – voided unrodded zero xenon conditions for T_m below 286 C.

Cold-HZP VOID Zero xenon (E = 0 MWd/kgHM)							
			K-infinite				
Mod. Temp. [C]	Void [%]	Dens. [g/cm3]	Trans. XS	Static XS	Reference	Err. Trans. XS [pcm]	Err. Static XS [pcm]
100	20	0.7668299	1.11665	1.10090	1.11596	69	-1506
	27	0.7000000	1.11191	1.09419	1.11209	-18	-1790
	60	0.3837138	1.08829	1.06805	1.08847	-18	-2042
	80	0.1921557	1.07107	1.04712	1.07123	-16	-2411
140	20	0.7413393	1.10963	1.09684	1.10951	12	-1267
	24.5	0.7000000	1.10685	1.09290	1.10702	-17	-1412
	60	0.3716520	1.08171	1.06545	1.08193	-22	-1648
	80	0.1868084	1.06484	1.04512	1.06501	-17	-1989
220	16.9	0.7000000	1.09725	1.09050	1.09743	-18	-693
	20	0.6745863	1.09555	1.08820	1.09570	-15	-750
	60	0.3430969	1.06804	1.05989	1.06820	-16	-831
	80	0.1773522	1.05177	1.04147	1.05183	-6	-1036
260	11	0.7000000	1.0927	1.08938	1.09283	-13	-345
	20	0.6317918	1.0878	1.08346	1.08794	-14	-448
	60	0.3277475	1.06063	1.05703	1.06073	-10	-370
	80	0.1757253	1.04474	1.04006	1.04475	-1	-469

Table X. Comparison of k-infinity between CROSS and PHOENIX4 using different sets of cross sections at 12 MWd/kg – voided unrodded equilibrium xenon conditions for T_m below 286 C.

Cold-HZP VOID Eq. xenon (E = 12 MWd/kgHM)							
			K-infinite				
Mod. Temp. [C]	Void [%]	Dens. [g/cm3]	Trans. XS	Static XS	Reference	Err. Trans. XS [pcm]	Err. Static XS [pcm]
100	20	0.7668299	1.21338	1.22498	1.21186	152	1312
	27	0.7000000	1.21015	1.22291	1.20994	21	1297
	60	0.3837138	1.19704	1.19656	1.19572	132	84
	80	0.1921557	1.18695	1.17367	1.18462	233	-1095
140	20	0.7413393	1.21382	1.22306	1.21305	77	1001
	24.5	0.7000000	1.21188	1.22152	1.21166	22	986
	60	0.3716520	1.19639	1.19373	1.19512	127	-139
	80	0.1868084	1.18518	1.17135	1.18302	216	-1167
220	16.9	0.7000000	1.21491	1.21893	1.21487	4	406
	20	0.6745863	1.21375	1.21755	1.21365	10	390
	60	0.3430969	1.19190	1.18763	1.19110	80	-347
	80	0.1773522	1.17763	1.16708	1.17616	147	-908
260	11	0.7000000	1.21600	1.21771	1.21616	-16	155
	20	0.6317918	1.21210	1.21342	1.21219	-9	123
	60	0.3277475	1.18727	1.18445	1.18695	32	-250
	80	0.1757253	1.17114	1.16542	1.17041	73	-499

4. CONCLUSIONS

In this paper the necessary cross section parameterization for BWR transient applications has been considered. The main objective has been to find a cross section representation model that provides complete transparency to users with regard to both steady-state and transient core analyses.

It has been demonstrated by means of simple 2D unit assembly test cases that if coolant boiling occurs below hot core operation conditions it will be necessary to include both the coolant density and the moderator temperature as explicit state parameters in the cross section tabulation. On the other hand, if no such boiling occurs at these intermediate moderator temperatures during the considered transient, no such decoupling of the moderator temperature from the coolant density is necessary with regard to the nodal cross section dependence on these state parameters.

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