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ABSTRACT

The VENUS PWR Mockup Experiment is an important component of the Nuclear Regulatory Commission's program goal of benchmarking reactor pressure vessel (RPV) fluence calculations in order to determine the accuracy to which RPV fluence can be computed. Of particular concern in this experiment is the accuracy of the source calculation near the core-baffle interface, which is the important region for contributing to RPV fluence.

Results indicate that the calculated neutron source distribution within the VENUS core agrees with the experimental measured values with an average error of less that 3%, except at the baffle corner, where the error is about 6%. Better agreement with the measured fission distribution was obtained with a detailed space-dependent cross-section weighting procedure for thermal cross sections near the core-baffle interface region. The maximum error introduced into the predicted RPV fluence due to source errors should be on the order of 5%.

INTRODUCTION

Radiation embrittlement of reactor pressure vessels (RPV) has recently been a concern of the nuclear industry because of the possibility that rapid cooling could lead to the failure of a brittle vessel. The U.S. Nuclear Regulatory Commission (USNRC) and the U.S. and European nuclear industries are currently conducting studies to determine the ability of PWR vessels to withstand severe thermal shocks without compromising their integrity. One of the major components of this research consists of benchmarking RPV fluence determination methods, since the RPV fluence is a determining factor in the degree of radiation embrittlement. An important part of the on-going RPV benchmark studies is called the "VENUS PWR Engineering Mockup Experiment."

Earlier benchmark experiments have focused on validating the accuracy of ex-core transport calculations to predict neutron fluence; however, they did not address the problem of determining the core fission source distribution which drives the RPV fluence calculation. Of particular concern is the accuracy of the source calculation near the core-baffle interface which is the important region for contributing to RPV fluence. The PWR Engineering Mockup Experiment was designed primarily to address this problem. The experimental work is being performed by CEN/SCK (Centre d'Etudie de l'Energie Nucleaire/Studie Centrum voor Kernenergie) at the VENUS Critical Facility in Mol, Belgium, while the calculational study is being done by both Mol and by the Louisiana State University Nuclear Science Center under subcontract to the Oak Ridge National Laboratory (ORNL).

The primary objective of this study is to compute the VENUS core neutron source distribution and compare with measured values to contribute to USNRC's program goal of validating RPV fluence calculations. The calculated fission source is then used as a fixed source for dosimeter calcul tions in order to ascertain the expected accuracy to which fluence can be predicted from a core eigenvalue calculation coupled with ex-core transport calculations.

EXPERIMENTAL CONFIGURATIONS

The PWR Benchmark Configuration in the VENUS Critical Facility is shown in Fig. 1. The central portion of the geometry is water, surrounded by a 2.858-cm thick inner steel baffle. The inner core zone in the immediate vicinity of the inner baffle contains 752 zircaloy-clad 3.3% enriched fuel cells, with 48 pyrex rods interspersed among them. The outer core zone contains 1800 steel-clad, 4.0%-enriched fuel cells. The core itself is surrounded by a 2.858-cm thick outer steel baffle, a water reflector, a 4.972-cm thick steel core barrel, a water gap, a neutron pad, and the reactor pool. (The neutron pad is not shown in Fig. 1.)

The configuration shown in Fig. 1 was selected by Mol as the core loading best suited for the realization of the required measurements in the fuel zones, reflector, barrel, and up to the

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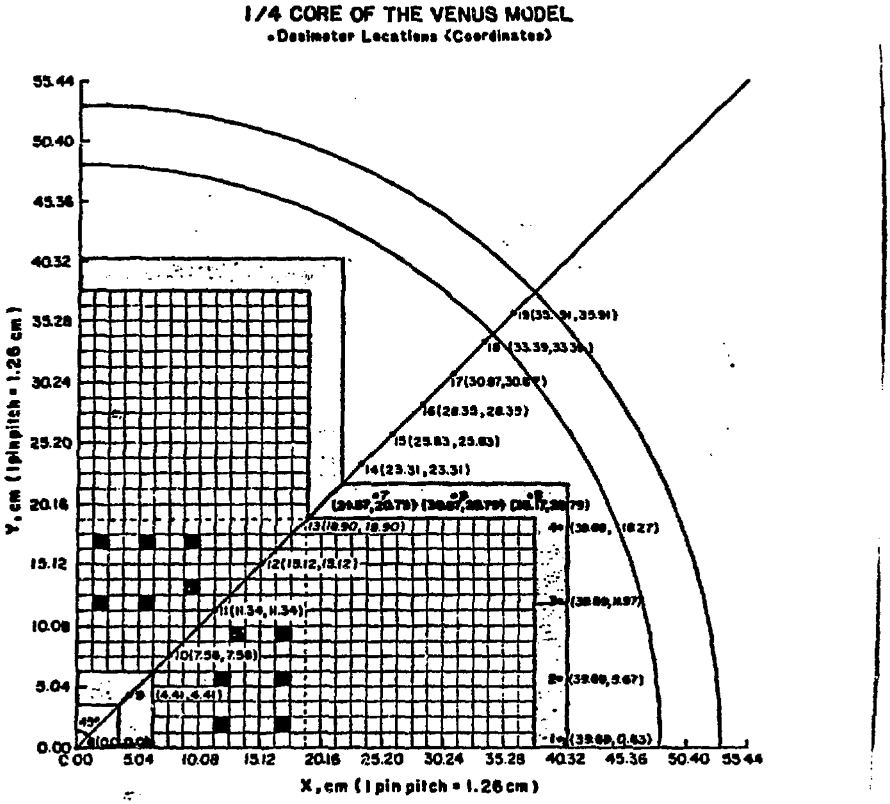

Fig. 1. One-fourth core of the VENUS model.

neutron pad. The distribution of pyrex rods in the inner zone of the core permits criticality without boron in water, and it shifts the power peak towards the core edges, thereby improving the core power distribution for the ex-core measurement. Details of the VENUS experiment configuration can be found in Ref. 3.

CALCULATIONAL METHODOLOGY

A 218-group cross section set based on ENDF/B-IV data (4) was used as the fine-group library. The AMPX (5) modular system was used for all the cross-section processing calculations. Resonance self-shielding was done with the Nordheim integral method as in the NITAWL (6) code, and the cell homogenization was performed with fluxes obtained from one-dimensional discrete ordinates calculations (XSDRN-PM) (7) of the 3 and 4% fuel cells and the pyrex-rod cell. The resulting 218-group resonance-shielded, cell-averaged cross sections were collapsed to 10 broad groups using zone-averaged fluxes computed for a one-dimensional model of the VENUS core.

The DOT IV (8) discrete ordinates transport theory code was used to perform a two-dimensional X-Y calculation of the critical eigenvalue. This calculation used the 10-group cross-section library discussed in the previous paragraph and was performed with a P_3 Legendre expansion of the cross sections and an S_8 quadrature set. The weighted flux differencing scheme was used, and the calculation was accelerated with the diffusion acceleration option. The VENUS model corresponded to the reactor midplane, extending radially past the core barrel. Axial leakage was treated with a buckling approximation using a single B^2 value obtained experimentally.

The space-dependent fission rate obtained in the 10-group eigenvalue calculation is compared to experimental results in the next section. Based on the experimentally measured total fission rate, the neutron source distribution found in this calculation was normalized appropriately and was input as the fixed source in a second DOT X-Y calculation which used the 56-group ELXSIR cross-section library (9). The calculation provided the various dosimeter activities which are discussed in the following section. The 56-group cross sections rather than the 10-group values were used for the dosimeter calculations because the dosimeter reactions have relatively high energy thresholds, and the 10-group structure is too coarse in this energy range.

RESULTS

While performing the 218-group one-dimensional transport calculations, it was observed that the thermal neutron spectrum significantly hardens near the interface of the core and the outer baffle (10). The rapid change in the thermal spectrum within the last 2 cm of the core was found to have a noticeable effect on the collapsed thermal cross-section values used in the two-dimensional calculations. In order to account for the spatial variation in the thermal cross sections, it was necessary to use a separately weighted set of collapsed cross sections at approximately every one-quarter centimeter within the last 2.52 cm (i.e., 2 cell widths) of the core. The spatial variation in the collapsed U-235 thermal fission cross section is shown in Table 1. The value for $^{235}U\ \sigma_f$ varies about 7% over this distance.

The effective multiplication factor for the two-dimensional X-Y calculations was determined to be $k_{eff} = 0.996$. This value was underpredicted by about one-half of one percent. The low value of k_{eff} is consistent with other LWR lattice studies which show that the ENDF/B-IV cross sections tend to underestimate the eigenvalue due to excessive U-238 capture estimates.

A comparison of the calculated and measured relative power distribution of the VENUS model is shown in Fig. 2 (10). The average agreement between calculation and experiment is within 3% error, with an uncertainty of about 1.5% in the measurements. The worst agreement has an error of 6.5%, and it occurs in a cell near the baffle corner. Disagreements of up to 3% can also be found at locations near the pyrex rods. The error introduced into the computed RPV fluence by these source discrepancies should be on the order of 5% or less.

A comparison of the calculated and measured dosimeter values is shown in Table 2. At most locations, the agreement between calculation and experiment is very good - nearly always within 10% and much better at many locations. The ^{237}Np results are an exception; the computed values are 10-17% higher than measured values in the core, and up to 30% lower in the water gap between the outer baffle and the barrel. There is also an odd value for ^{235}U in the inner baffle for which the measured value is about 35% higher than our calculation. We suspect an experimental problem here.

CONCLUSIONS

The space-dependent neutron fission rate in the VENUS core, particularly at the core periphery, can be accurately calculated with discrete ordinates transport theory. A high degree of accuracy can be obtained by using space-dependent cross-section weighting for ^{235}U thermal fission in the important outer baffle-core interface region, although this procedure may not be practical for power reactor analysis.

Comparison of calculation with measured relative power distribution indicates that the spatial neutron source can be computed to an accuracy of within 6% error near the important core-baffle region and an average agreement of about 3% error for the whole in-core area.

When the computed neutron source distribution is used in transport calculations of dosimeter activation, the results are found to agree very well with experimental measurements, except at a few locations at which the experimental values are suspect. The agreement for most dosimeters is usually better than 7%. The ^{237}Np values only agree to 10-17% within the core, however.

Overall, it is concluded that determination of the core source with a calculational procedure of this type is an adequate approach for pressure vessel fluence calculations.

Table 1. Variation of U-235 thermal fission cross section*

Distance from outer baffle (cm)	U-235 group 10/10 fission cross section (barns)
0.252	250.35
0.504	255.79
0.756	259.64
1.008	262.40
1.260	264.55
1.512	266.28
2.520	269.41
2.52-18.90	278.80

*These are collapsed values and are applicable to the 4.0% fuel region only.

Table 2. C/E values for dosimeter results

Dosimeter location (see Fig. 1)	^{115}In(n,n')	^{58}Ni(n,p)	^{238}U(n,f)	^{237}Np(n,f)	^{235}U(n,f)
1	0.99	1.08	1.06	1.07	----
2	0.99	1.08	1.08	----	----
3	0.98	1.05	1.05	1.09	0.97
4	0.98	1.03	----	1.10	0.98
5	0.98	1.02	1.07	1.10	----
6	0.96	1.07	1.04	1.10	----
7	0.96	1.05	1.06	1.08	0.95
8	0.98	1.07	1.01	0.97	----
9	0.99	1.10	1.08	1.17	0.62
10	----	----	----	1.14	----
11	----	----	----	1.12	----
12	----	----	----	1.14	----
13	----	----	----	1.13	----
14	----	----	----	1.01	----
15	----	----	----	0.90	----
16	----	----	----	0.70	----
17	----	----	----	0.82	----
18	----	----	----	0.76	----
19	0.93	0.83	0.98	1.01	----

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OUTER BAFFLE

									1.023	0.988	0.999	1.019	1.010	1.009	1.023	0.993	1.004	1.012	0.985	1.002	1.033	1.056	1.065	1.053
								1.025	1.036	1.015	1.007	0.992	1.009	1.004	0.999	0.989	1.006	0.998	1.000	1.020	1.002	1.046	1.050	1.033
							0.990	0.997	0.994	1.009	1.015	1.005	0.999	0.995	0.986	0.985	0.996	1.007	1.006	1.007	1.025	1.013	1.054	0.993
						0.993	1.009	0.995	0.999	1.011	0.989	0.998	0.981	0.981	0.984	0.986	0.972	0.989	0.988	1.006	0.991	1.024	1.005	0.985
					1.005	0.986	1.020	0.998	1.002	0.997	0.984	0.981	0.982	0.975	0.988	0.977	0.985	0.981	0.994	0.993	1.000	1.002	1.002	1.000
				0.975	0.978	0.976	0.990	1.013	1.005	1.005	0.986	0.986	0.986	0.988	0.987	0.991	0.977	0.988	0.981	0.997	1.005	1.015	0.992	1.000
			0.968	0.979	0.983	0.971	0.991	0.991	1.000	0.998	0.985	0.980	0.989	0.992	0.988	0.986	0.997	0.997	0.994	0.984	1.012	1.026	1.021	0.995
		0.980	0.990	0.981	⊠	0.976	1.022	⊠	1.018	1.006	0.995	0.992	0.992	1.004	0.995	1.000	1.000	1.001	0.993	0.996	1.004	1.027	1.022	0.998
	1.004	0.978	0.971	0.961	0.976	0.964	0.990	1.015	1.001	1.025	1.004	1.007	1.003	1.014	1.009	1.003	1.008	1.005	1.001	1.000	1.017	1.037	1.029	1.007
0.974	0.990	0.983	0.980	0.977	0.994	0.982	1.006	1.017	1.016	1.018	1.004	1.008	1.014	1.007	1.005	1.004	1.011	1.006	1.003	1.002	1.000	1.022	1.051	1.006
1.029	1.013	0.981	0.966	⊠	0.973	0.988	0.982	⊠	1.031	1.018	1.012	1.012	1.015	1.008	1.008	1.004	1.008	0.999	1.000	1.001	1.010	1.023	1.027	1.008
1.012	1.004	0.986	0.996	0.979	0.997	0.989	1.014	1.024	1.020	1.018	1.007	1.009	1.012	1.007	0.997	1.003	1.003	1.016	0.998	1.003	1.014	1.026	1.016	1.016
0.999	0.987	0.963	0.984	0.977	0.991	0.975	1.018	1.024	1.015	1.000	0.999	1.003	1.012	1.002	0.997	0.983	0.992	1.003	1.006	1.001	1.013	1.023	1.031	1.008
1.006	0.991	0.982	0.984	⊠	0.977	0.987	0.984	⊠	1.040	1.014	1.004	1.009	1.008	1.009	1.001	1.001	1.001	1.001	1.002	1.013	1.004	1.034	1.022	1.020
1.016	0.989	0.971	0.996	0.980	1.014	0.993	1.002	1.025	1.027	1.006	1.006	1.002	1.007	1.007	0.989	1.003	1.005	1.010	1.001	1.009	1.012	1.028	1.026	1.026

INNER BAFFLE

OUTER BAFFLE

⊠ - PYREX RODS

Fig. 2. Comparison of calculated and measured (C/E) relative power distribution.

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