NEUTRONIC FLUX CALCULATION
FOR THE INNER WALL OF A TYPICAL BWR VESSEL

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ABSTRACT
The paper discusses the methodology used to develop the calculational methods for estimating the fast flux, and thus the embrittlement, in the pressure vessel of a BWR reactor. The two dimensional Sn transport code DORT was used for the calculations. A three-dimensional approximation of the flux was obtained by "synthesizing" fluxes obtained from two-dimensional transport calculations in R-theta and RZ cylindrical geometries along with a single one-dimensional radial calculation. The experimental value of the flux was determined from measurements of gamma emission rates from Fe and Cu dosimeters extracted from the reactor vessel. The reaction rates were used as input to the SAND II computer code to calculate the experimental values of the fast flux. The calculational results of the fast flux values were compared with the experimentally determined values of the fast flux in order to verify the calculations. The two results agreed within 4.2 percent. The work was preformed as part of the embrittlement surveillance program of the Laguna Verde BWR and will be used for prediction of future radiation damage.

1. INTRODUCTION

A reactor vessel is continuously being bombarded by radiation. The effects of such radiation results in embrittlement of the vessel materials, specifically in the center core region or beltline. The norm for licensing a nuclear power plant requires that the reactor coolant pressure boundary be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing and possible accident conditions, the boundary behaves in a nonbrittle manner and also that the probability of rapidly propagating fracture is minimized. This requires the knowledge of changes in the fracture toughness of the reactor vessel caused by neutron irradiation. A surveillance program for a BWR reactor vessel provides the assurance that a brittle fracture is prevented [1].

This paper discusses part of the development of a surveillance program for the Laguna Verde BWR reactor. To develop the program it was necessary to establish experimental benchmarks and then develop the calculational methodology necessary to reproduce the experimental results. The calculational methods may then be used, within limits, to predict future damage.

In the first section of this paper, the methodology and results of experiments used to determine the flux in the Laguna Verde reactor inner vessel wall will be discussed. The experimental value of the flux was determined from measurements of gamma emission rates from Fe and Cu dosimeters extracted from the reactor vessel. The neutron flux was calculated from the gamma activity measurements using the SAND II [2] computer code. These measurements are reported in the paper and are used as the benchmark for establishing the computational methods.

The next section will discuss the computational methods used to reproduce the experimental results. The choice of neutronic methods chosen to reproduce the experimental flux was precipitated by the desire to run many future survey calculations for prediction of the vessel flux in relatively short times. Thus the transport code DORT-4 [3], a two-dimensional Sn code, was chosen for the calculations.
The results of the comparison and conclusions will be presented in the final section of the paper. The results of the calculations demonstrate good agreement between the experimental and calculational results, thus validating surveillance program developed.

2. EXPERIMENTAL METHODS AND RESULTS

Measurements of gamma emission rates from Fe and Cu dosimeters that have been exposed to the neutron fluence on a reactor vessel can be used to determine the fluence. The dosimeter's activity is related to the total neutron exposure, thus the absolute neutron fluence on the pressure vessel wall may be determined [4]. The fluence may be used then to determine the reactor material's embrittlement caused by the neutron bombardment.

As part of the Laguna Verde reactor vessel materials surveillance program, Fe and Cu dosimeters were placed adjacent to the pressure vessel wall [5]. These flux wire dosimeters were positioned at 30° azimuth and were removed after concluding the first operation cycle. They were then subjected to analysis. The neutron flux is proportional to the fission rate and is nearly proportional to the reactor thermal power. Therefore, the results from the neutron flux wire dosimeters were used to establish a point in the relation between vessel neutron fluence and thermal power. Extrapolating from this point, predictions can be made to obtain the total neutron exposure during the remaining reactor operational lifetime.

As noted, the neutron flux wire dosimeters were extracted in the first operational cycle of the Laguna Verde nuclear plant. This corresponds to 692 working days at a 0.667 average capacity, or 412 days at full power or 1.26 effective full power years. Three copper and three iron dosimeters were measured for the gamma activity for the determination of the absolute neutron flux. The experiment was used to determine that neutron fluence with energy greater than 1 MeV.

The reactions of interest in the iron and copper dosimeters when bombarded with a neutron flux are: $^{54}Fe(n,p)\ ^{54}Mn$ and $^{63}Cu(n,\alpha)^{60}Co$ [4,5,6]. The dosimeter results and the SAND II code was used to approximate the fast flux. For a 100 percent capacity factor, the neutron flux (E > 1 MeV) was determined to have a value of $2.71\times10^{9} \pm 3.28\times10^{8}$ n/cm^2-s [7].

3. CALCULATIONAL METHODOLOGY

In this section, the development the calculational methodology necessary to reproduce the experimental results are discussed. The choice of neutronic methods chosen to reproduce the experimental flux was precipitated by the desire to run many future survey calculations for prediction of the vessel flux in relatively short times. If the methodology developed can reproduce the experimental results, the effect different fuel configurations on the embrittlement damage of the pressure vessel could then be, within limits, predicted.

For this work, a three-dimensional approximation of the flux in the pressure vessel of a BWR was obtained by "synthesizing" fluxes obtained from two-dimensional transport calculations in R-theta and RZ cylindrical geometries along with a single one-dimensional radial calculation. These three sets of results can be combined using Equation 1, obtained from Reference 8, to obtain a three-dimensional approximation of the flux, $\Phi(R,\Theta,Z)$, in R-theta-Z coordinates. The expression for approximating the three dimensional flux is

$$\Phi(R,\Theta,Z) = \frac{\phi_{R,\Theta}(R,\Theta) \times \phi_{R,Z}(R,Z)}{\phi_R(R)} \qquad (1)$$

where:

$\phi_{R,\Theta}$ = "R-theta channel" flux, obtained from a 2D transport calculation in R-theta geometry,

$\phi_{R,Z}$ = "RZ channel" flux, obtained from a 2D transport calculation in RZ geometry, and

ϕ_R = "R channel" flux, obtained from a 1D transport calculation in R geometry; corresponding to a radial traverse along the core midplane in the RZ model.

The two-dimensional transport code, DORT-4, was used to estimate the R-theta and the RZ fluxes. DORT-4 was also used to estimate the one dimensional radial flux. The angular quadrature used was an S_8 order.

A six group cross section library was used in the calculations. The library was collapsed from the VITAMIN-B6 [9] fine-group library using the methodology described in Reference 10. VITAMIN-B6 is derived from ENDF/B-VI nuclear data. This fine-group library, consists of 199 neutron energy groups and 42 gamma-ray energy groups and contains data for 123 nuclides. The six energy groups structure is noted in Table 1.

The R-theta coordinate system calculational model corresponds to a slice through an infinitely tall cylinder, in which the neutron source and material composition (and hence the flux) can vary azimuthally and radially. The irregularly shaped core boundary must be represented by different radii, as a function of theta. This causes the out-core flux to vary significantly as a function of theta for a fixed radius, since the amount of water between the core boundary and the shroud radius changes as theta changes. Figure 1 shows the geometry which was converted into the R-theta coordinates used in the transport calculations.

Table 1.Neutron energy groups structure

Group	Upper energy, (eV)	Lower energy, (eV)
1	1.9640×10^7	1.0000×10^7
2	1.0000×10^7	1.0026×10^6
3	1.0026×10^6	9.1188×10^3
4	9.1188×10^3	5.0435
5	5.0435	0.41399
6	0.41399	5.0000×10^{-4}

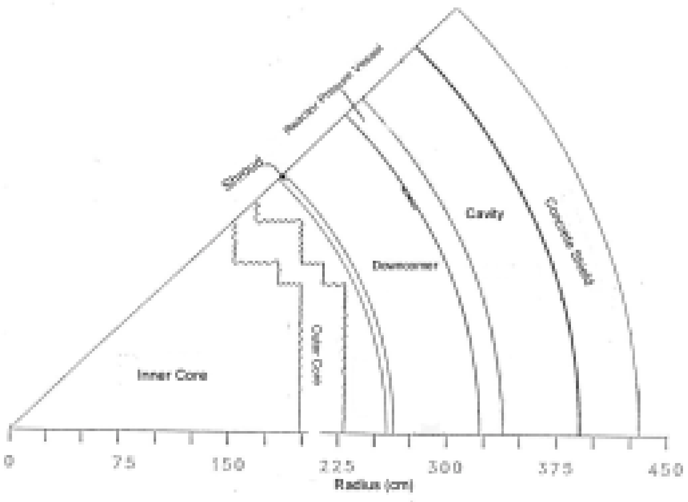

Figure 1. Sectional Cross Section of Reactor Core

The R-theta calculation provides the azimuthal shape of the out-core flux for elevations within the active core height. A void fraction of about 50% was used in the R-theta model, which corresponds roughly to the midplane void fraction. The impact of the axial variation in the void fraction is accounted for in the RZ transport calculation, which is discussed next.

The calculational model used in the RZ calculations contained two separate regions. These two major regions, the upper and lower models are shown in Figures 2 and 3 respectively. In order to reduce the size of the problem, only one half of the reactor is modeled in each RZ calculation. The origin of the Z axis used in RZ transport calculations is located at the center of the active core; hence elevations above the reactor midplane have positive Z coordinates, while those below the midplane have negative values.

The variation of void density with core height that occurs in a BWR and the variation in water and steam fraction were modeled as seven zones with different void fractions. The values used were based on a typical BWR axial void distribution. Control rod positions were assumed to be those of a typical BWR configuration. To obtain a sufficiently accurate neutron source distribution, the power shape was calculated across the outer fuel assemblies in order to predict fluxes in outer regions near the vessel.

The 1D radial model (R) is defined by a radial traverse of either RZ model at the core midplane (Z=0). It includes the homogenized jet pumps in the model.

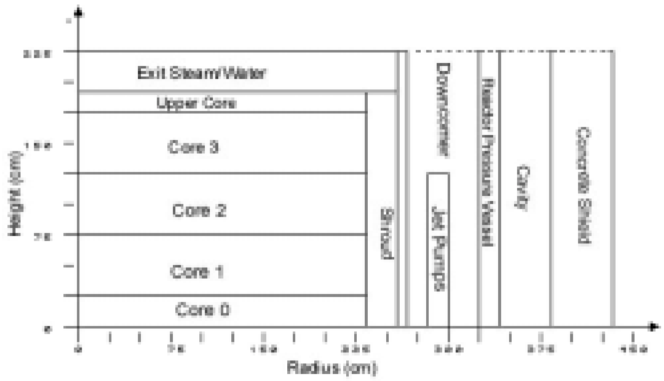

Figure 2. Layout of the upper section of BWR, RZ plane (origin of Z coordinate is the core midplane.)

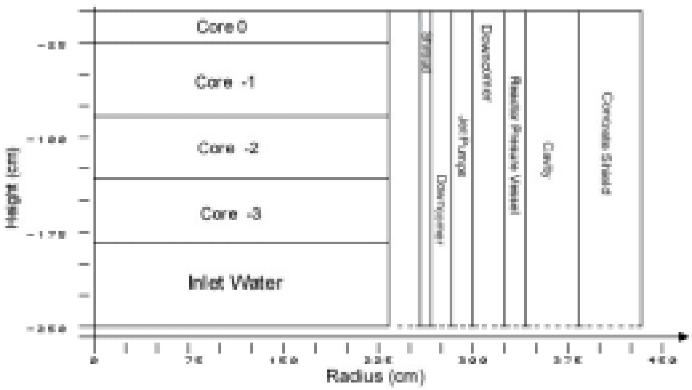

Figure 3. Layout of the lower of representative BWR RZ plane. (Origin of the Z coordinate is the core midplane.)

Although two-dimensional RZ geometry is able to correctly represent the radial and axial variation in the rector dimensions and compositions, it must be assumed that the components are axial-symmetrical. For true cylindrical components like the shroud, vessel, etc. this assumption is strictly correct, however, the outer boundary of the core region can not be exactly modeled in RZ geometry, since it is not axial-symmetrical (i.e,. the core radius is a function of theta). For the effective radius in our RZ and R models we have used an average radius which preserves the total core volume.

Material compositions in the various zones of the DORT models are homogenized mixtures of water, stainless steel, core, etc, based on volume fractions given in the plant engineering drawings. A typical axial void distribution was used to determine the core water density as a function of elevation in active core: seven zones with different void fractions were represented in the DORT RZ models to account for variation in the water density from the core inlet to the outlet. The core inlet water is slightly sub-cooled, while the exit void fraction at the top of the core is set to 70%. The water/steam mixture within the shroud dome above the core was assumed to be at 65% void. The water below the core is sub-cooled at 1000 psi, and has a density of about 0.74 g/cc. The control rod insertion pattern is represented by 12 “deep rods” which are fully inserted; 12 “shallow” rods which are inserted up to an elevation of about 109 cm. above the bottom of the core; and the remainder of the rods are assumed to be fully withdrawn. The B_4C absorber in the blades was uniformly homogenized within the volumes containing the control rods. Table 2 lists the compositions of all materials used in the DORT transport models.

Table 2. Material Compositions used in DORT Calculations

Material/Region	Material/Nuclide	Atom Densities (atom/b-cm)
Fuel	O	1.0415×10^{-2}
	^{235}U	9.4546×10^{-5}
	^{238}U	5.1120×10^{-3}
	Zr	5.7000×10^{-3}
Sat. H_2O	H	4.9284×10^{-2}
	O	$2.4642 \ X \ 10^{-2}$
Downcomer Coolant (DC. H_2O)	H	5.0455×10^{-2}
	O	2.5228×10^{-2}
65% Void Water	Sat H_2O.	0.35
Carbon Steel (CS)	Cr	1.2700×10^{-4}
	Mn	1.1200×10^{-3}
	Ni	4.4400×10^{-4}
	Fe	8.1900×10^{-2}
	C	9.8100×10^{-4}
	Si	3.7100×10^{-4}
Stainless Steel (SS)	Cr	1.7400×10^{-2}
	Mn	1.5200×10^{-3}
	Ni	8.5500×10^{-3}
	Fe	5.8300×10^{-2}
	C	2.3700×10^{-4}
	Si	8.9300×10^{-4}
Homogenized Control Rod (CR) Absorber per rod in region	^{10}B	6.9404×10^{-7}
	C	8.7647×10^{-7}
Jetpump	H	4.1394×10^{-2}
	O	2.0697×10^{-2}
	Cr	2.7867×10^{-3}
	Fe	1.1456×10^{-2}
	Ni	1.2385×10^{-3}

Table 2 continued

Material/Region	Material/Nuclide	Multplier
Core -3	Fuel Sat. H_2O C. R:	1.0000 0.6132 24.0
Core -2	Fuel Sat. H_2O C. R	1.0000 0.5630 24.0
Core -1	Fuel Sat. H_2O C. R	1.0000 0.4285 12.0
Core 0	Fuel Sat. H_2O C. R	1.0000 0.3303 12.0
Core 1	Fuel Sat. H_2O C. R..	1.0000 0.2680 12.0
Core 2	Fuel Sat. H_2O. C. R.	1.0000 0.2256 12.0
Core 3	Fuel Sat. H_2O C. R.	1.0000 0.2087 12.0
Upper Core	Core 3 S. S.	0.8750 0.1250
Reflector	Sat. H_2O	1.0000
Shroud	S.S.	1.0000
Downcomer	DC H_2O	1.0000
RPV Wall	CS	1.0000
Exit H_2O – 65% Void	65% Void H_2O	1.0000

4. RESULTS

It was found that the axial elevation at which the thermal flux peaks changes as function of radius. Near the core centreline the thermal flux peaks about 1.345 cm below the reactor midplane, but at radii beyond the active core the peak shifts above the midplane. The peak values the fast (1 MeV < E < 10 MeV) and thermal (E < 0.414 eV) fluxes where the maximum axial fluxes occur at the reactor vessel (R=321.3 cm) are equal 8.64 x 10^9 n/cm^2-s and 5.34 x 10^{10} n/cm^2-s, respectively (Theta = 44.82 ^{o} and 38.70 ^{o}, respectively). The peak fast flux at 1/4T location of the RPV is equal to 7.28 x 10^9 n/cm^2-s. The fast and thermal fluxes at the surveillance capsule position (Theta = 32.09 ^{o}, R = 321.3 cm) are calculated to be equal to 2.83 x 10^9 and 4.24 x 10^{10} n/cm^2 s, respectively. A comparison of the experimental to the calculational results is given in Table 3.

Table 3.
A Comparsion of the Calculational Fast Flux
to the Experimental Fast Flux at the Foil Location

Experimental Fast (EF) Flux, n/cm^2-s (E > 1 MeV)	Caclulational Fast (CF) Flux, n/cm^2-s (E > 1 MeV)	Difference, Per Cent [(CF-EF)/CF]x100
2.71×10^9	2.83×10^9	4.2

5. CONCLUSION

The results indicate that difference between the calculational and experimental fast flux, 4.2%, on the pressure vessel wall is sufficiently small enough that the methodology may be used for survey calculations for predicting embrittlement. However, once a fuel pattern has been established for use, three dimensional calculations will be made to obtain a better approximation for the flux at regions beyond the beltline.

6. REFERENCES

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