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RECENT PROGRESS AND DEVELOPMENTS IN LWR-PV CALCULATIONAL
METHODOLOGY*

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ABSTRACT

New and improved techniques for calculating belt-line surveillance activities and pressure vessel fluences with reduced uncertainties have recently been developed. These techniques involve the combining of monitored in-core power data with diffusion theory calculated pin-by-pin data to yield absolute source distributions in R-6 and R-Z geometries suitable for discrete ordinate transport calculations. Effects of finite core height, whenever necessary, can be considered by the use of a three-dimensional fluence rate synthesis procedure. The effects of a time-dependent spatial source distribution may be readily evaluated by applying the concept of the adjoint function, and simplifying the procedure to such a degree that only one forward and one adjoint calculation are required to yield all the dosimeter activities for all beltline surveillance locations at once. The addition of several more adjoint calculations using various group fluence rates as responses is all that is needed to determine all the pressure vessel group fluences for all beltline locations for any arbitrary source distribution.

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INTRODUCTION

In enlarging the original concept of the LEPRICON adjustment procedure (1,2,3) to include methods for obtaining a priori values of calculated surveillance dosimeter activities and pressure vessel fluence rates in PWR systems (4,5), as well as in applying some of these methods to the analysis of experiments performed in reactor-like environments (6,7,8), new and improved techniques have been developed to handle various aspects of the general transport process.

ABSOLUTE NEUTRON SOURCE DISTRIBUTIONS

For analysis of PWR surveillance dosimetry, it has been found adequate in the analysis of the Arkansas Nuclear One-Unit I reactor (ANO-1) to use in-core assembly power data determined from self-powered neutron detectors (SPND) in conjunction with diffusion theory calculations that provide reasonable intra-assembly power distributions for the peripheral assemblies. Thus,

$$P(x,y,z,m) = P(z,m)[p(x,y,m) / \int_{V_m} p(x,y,m)dV], \quad (1)$$

where $P(x,y,z,m)$ is the power spatial distribution for peripheral assembly m , $P(z,m)$ is the axial power distribution as measured by SPND's in assembly m , and $p(x,y,m)$ is the diffusion theory calculated horizontal midplane power distribution for assembly m . By defining Eq. (1) to hold over some coarse time interval Δt_j and multiplying it by $F_i(\Delta t_j)$, the "instantaneous" fraction of the SPND power of duration $\Delta t_i, \Delta t_i \in \Delta t_j$, which is extracted from the power history and reflects power level swings during operation, a suitable approximation to the peripheral power distribution as a function of time may be written as

$$P(x,y,z,m,\Delta t_j) = F_i(\Delta t_j)P_j(z,m,\Delta t_j)[p_j(x,y,m) / \int_{V_m} p_j(x,y,m)dV]. \quad (2)$$

Throughout cycles 4 and 5 of this reactor, which have been the only ones analyzed so far, the virtually time-independent pin-by-pin peripheral assembly distributions were such that the core leakage averaged about 15-20% less than that from a flat distribution normalized to the same assembly powers. Thus, neglecting the fall-off in pin-by-pin power with increasing radius for the peripheral assemblies would result in typical overestimates of the order of 20% in calculated pressure vessel fluences. The conversion factor relating neutron production rate to the power expressed in Eq. (2) is burn-up dependent because of different values for ^{239}Pu and ^{235}U . The fission spectrum is also somewhat different for these two isotopes. Both of these effects tend to increase the high-energy fluence with time over that obtained if

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they were neglected, but their combined effect is probably significant (>5%) only for low leakage cores in which partially burned fuel is used in the peripheral assemblies.

EFFECTS OF THE FINITE CORE HEIGHT

These effects and their treatment have been discussed elsewhere (6,7,8,9) for near-midplane locations, but only corrections for PWR reactor cavity positions probably need to be considered. For the PCA and PSF experiments, however, even calculations at surveillance and pressure vessel midplane locations need to be corrected, since the core height is relatively small compared to detector distances from the core. For axial locations further than about one-fourth of the core height from the horizontal midplane, the simple fluence rate synthesis procedure given below begins to break down. The procedure is based on combining results from three transport calculations. For PWR applications,

$$\phi_g(R, \theta, Z) = \phi_g(R, \theta) \phi_g(R, Z) / \phi_g(R) \quad , \quad (3)$$

where

$$\phi_g(R, \theta) \text{ is the solution using } S(R, \theta) = \int_{-\infty}^{\infty} S(R, \theta, Z) dZ \quad , \quad (4)$$

$$\phi_g(R, Z) \text{ is the solution using } S(R, Z) = \int_0^{2\pi} S(R, \theta, Z) R d\theta \quad , \quad (5)$$

and

$$\phi_g(R) \text{ is the solution using } S(R) = \int_0^{2\pi} R d\theta \int_{-\infty}^{\infty} S(R, \theta, Z) dZ \quad . \quad (6)$$

In applications to ANO-1, and presumably many other reactors, the ratio $\phi_g(R, Z) / \phi_g(R)$ was assumed to be independent of the source distribution because all axial loadings were virtually identical.

EFFECTS OF TIME-DEPENDENT SPATIAL SOURCE DISTRIBUTIONS

Source spatial distributions in general are functions of time, and both intracycle and intercycle variations may be important as they can affect pressure vessel fluence. A method based on the concept of the adjoint function has been developed and simplified which allows one to obtain not only surveillance activities at the end of irradiation and pressure vessel fluences for any arbitrary source distribution without the time and expense of performing new transport calculations each time, but also to investigate effects on pressure vessel fluence alone from different core loadings and fuel shuffling schemes.

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Turning our attention first to calculating the dosimeter saturated activities $RR^d(t)$,

$$RR^d(t) = \int_{\text{detector}} d\vec{r} \int_0^{\infty} \phi(E, \vec{r}, t) \sigma^d(E) dE = \int_{\text{core}} d\vec{r} \int_0^{\infty} \phi_d^*(E, \vec{r}) S(E, \vec{r}, t) dE, \quad (7)$$

where ϕ is the "forward" calculated fluence rate, ϕ_d^* is the adjoint fluence rate using σ^d , the reaction cross section for dosimeter d , as the adjoint source, and S is the core source.

$$\text{Since } S(E, \vec{r}, t) = \chi(E) S(\vec{r}, t), \quad (8)$$

where χ is a normalized fission spectrum,

$$RR^d(t) = \int_{\text{core}} S(\vec{r}, t) d\vec{r} \int_0^{\infty} \chi(E) \phi_d^*(E, \vec{r}) dE. \quad (9)$$

We may rewrite Eq. (9) as

$$RR^d(t) = \sum_{k=1}^8 \int_0^{\infty} \chi(E) dE \int_{\text{octant}} \phi_d^*(k, \vec{r}, E) S(k, \vec{r}, t) d\vec{r}, \quad (10)$$

where k denotes a given octant of the core.

Most fuel loading patterns exhibit the octal symmetry property $S(k, \vec{r}, t) = S(\vec{r}, t)$ for all k , leading to

$$RR^d(t) = \sum_{k=1}^8 \int_0^{\infty} \chi(E) dE \int_{\text{octant}} \phi_d^*(k, \vec{r}, E) S(\vec{r}, t) d\vec{r}. \quad (11)$$

Equation (11), requiring a model of the complete core, may be greatly simplified if the adjoint function is calculated modeling only the nearest octant of the core together with imposing a perfectly reflecting boundary condition on the two limiting rays of the octant (see Fig. 1). Since, by symmetry, Fig. 1 is equivalent to no reflective boundary condition and eight symmetrically placed sources about the core (broken crosses in Fig. 1), one has, in terms of the reflected adjoint function $\phi_{d,R}^*$,

Fig. 1. Reflected Geometry Using One Octant of the Core

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$$\int_0^{\infty} \chi(E) dE \int_{\text{core}} \phi_{d,R}^*(\vec{r}, E) S(\vec{r}, t) d\vec{r} = 8RR^d(t), \quad (12)$$

and since all octants are identical,

$$\int_0^{\infty} \chi(E) dE \int_{\text{octant}} \phi_{d,R}^*(\vec{r}, E) S(\vec{r}, t) d\vec{r} = RR^d(t). \quad (13)$$

Thus, subtracting Eq. (13) from Eq. (11) and interchanging the order of octant summing and integration,

$$\int_0^{\infty} \chi(E) dE \int_{\text{octant}} \left\{ \sum_{k=1}^8 \phi_d^*(k, \vec{r}, E) - \phi_{d,R}^*(\vec{r}, E) \right\} S(\vec{r}, t) d\vec{r} = 0, \quad (14)$$

and since $S(\vec{r}, t)$, is arbitrary,

$$\sum_{k=1}^8 \phi_d^*(k, \vec{r}, E) \equiv \phi_{d,R}^*(\vec{r}, E). \quad (15)$$

Thus, although the adjoint geometry is asymmetric, the adjoint function readily calculated with reflective boundary conditions in octal geometry is the correct solution if the core loading is octally symmetric. The quantity $\chi(E)\phi_{d,R}^*(\vec{r}, E)$ is the Green's function for the response function $\sigma^d(E)$ and a particular location of the detector.

Figure 2 illustrates the behavior of the ratio of the Green's functions of $^{237}\text{Np}(n, f)$ relative to $^{63}\text{Cu}(n, \alpha)$ for surveillance dosimeters located near the region of peak pressure vessel fluence in ANO-1. The five shaded quarter assemblies represent regions of greatest contribution to the dosimeter activities, and the near constancy of this ratio, which involves two completely different energy intervals, is to be observed. Similar conclusions can be inferred from the distributions in Fig. 3, where the dosimeters are located in the ANO-1 reactor cavity. One may conclude from Figs. 2 and 3 that the similar spatial dependence of the two Green's functions over the contributing part of the core for two such disparate response functions implies, by Eq. (13), the independence of the relative reaction rates with source distribution. Hence the relative fluence rates at a given surveillance or pressure vessel location also tend to be independent of spatial source distribution. Only the intensity of the total fluence rate is affected by the source distribution; the spectrum at a given location remains virtually independent of the source.

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Fig. 2. Ratio of Weighted Source Importances of Quarter Peripheral Core Assemblies Summed Over All Octants, $^{237}\text{Np}(n,f)/^{63}\text{Cu}(n,\alpha)$, in Units of 10^3 . Detector at Surveillance Location in Downcomer.

Fig. 3. Ratio of Weighted Source Importances of Quarter Peripheral Core Assemblies Summed Over All Octants, $^{237}\text{Np}(n,f)/^{63}\text{Cu}(n,\alpha)$, in Units of 10^4 . Detector in Reactor Cavity.

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The constancy of the spectrum allows one to calculate all the dosimeter activities for a new source distribution by using only the one adjoint function and scaling the original dosimeter activities by the ratio of the new to the old activities obtained for the particular dosimeter whose response cross section was used in calculating the adjoint function. Thus the saturated activities $RR^d(t)$ and $RR^d(t_0)$ calculated from Eq. (13) using the sources $S(\bar{r}, t)$ and $S(\bar{r}, t_0)$, respectively may be used in conjunction with the activities $RR^d(t_0)$ obtained from a forward calculation performed earlier using the reference source $S(\bar{r}, t_0)$ to provide new saturated activities $RR^d(t)$ for all the dosimeters:

$$RR^d(t) = \{ [RR^d(t)]_{\text{adjoint}} / [RR^d(t_0)]_{\text{adjoint}} \} [RR^d(t_0)]_{\text{forward}} \quad (16)$$

As an illustration, we have calculated in the forward mode the activities in the ANO-1 cavity for each of two source distributions - one flat and the other modified by a pin-by-pin diffusion theory calculation. Taking the flat distribution as the new distribution and the pin-by-pin distribution as the original one, the new activities were calculated as well by scaling the original ones using each of two adjoint functions in Eq. (16). The results appear in Table 1.

Table 1. Comparison of Forward with Adjoint Scaled Saturated Activities in the ANO-1 Cavity for a Flat Intra-Assembly Source Distribution

Dosimeter	Using $^{63}\text{Cu}(n, \alpha)$ Adjoint	Using $^{237}\text{Np}(n, f)$ Adjoint	Forward	Discrepancy	
$^{63}\text{Cu}(n, \alpha)^{60}\text{Co}$	6.43-17†	6.74-17	6.43-17	0.0%	4.8%
$^{46}\text{Ti}(n, p)^{46}\text{Sc}$	8.85-16	9.29-16	8.92-16	1.0%	4.1%
$^{54}\text{Fe}(n, p)^{54}\text{Mn}$	4.86-15	5.10-15	4.97-15	2.3%	2.6%
$^{58}\text{Ni}(n, p)^{58}\text{Co}$	6.74-15	7.07-15	6.90-15	2.4%	2.5%
$^{238}\text{U}(n, f)\text{F.P.}$	2.83-14	2.98-14	2.93-14	3.5%	1.7%
$^{237}\text{Np}(n, f)\text{F.P.}$	4.92-13	5.17-13	5.17-13	5.1%	0.0%

†Read 6.43×10^{-17} reactions per second per atom, etc.

Table 1 indicates about the same error is incurred using either extreme adjoint function, but that using an intermediate one calculated from either the $^{54}\text{Fe}(n, p)$ or the $^{58}\text{Ni}(n, p)$ cross section should result in no worse than about a 2.5% error.

Even though the pressure spectrum changes with detector location throughout the pressure vessel for a given source, no changes in these relative spectra are induced by changes in the source; i.e., these spectra are determined strictly by the neutron transport

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properties of the downcomer and pressure vessel. Hence, one can use Eq. (16) as well to use scaling factors at one location to obtain activities at another, as is illustrated in Table 2.

Table 2. Comparison of Forward with Downcomer Adjoint Scaled Saturated Activities in the ANO-1 Cavity for a Flat Intra-Assembly Source Distribution

Dosimeter	Using $^{63}\text{Cu}(n,\alpha)$ Adjoint [†]	Using $^{237}\text{Np}(n,f)$ Adjoint [†]	Forward	Discrepancy	
$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$	6.18-17*	6.46-17	6.11-17	1.1%	5.8%
$^{46}\text{Ti}(n,p)^{46}\text{Sc}$	8.48-16	8.87-16	8.45-16	0.4%	4.9%
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	4.67-15	4.88-15	4.73-15	1.2%	3.3%
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	6.47-15	6.76-15	6.57-15	1.6%	2.9%
$^{238}\text{U}(n,f)\text{F.P.}$	2.71-14	2.84-14	2.80-14	3.1%	1.5%
$^{237}\text{Np}(n,f)\text{F.P.}$	4.77-13	4.98-13	4.96-13	4.1%	0.4%

$$^*(\text{RR}^d)_{\text{FLAT,CAVITY}} = \frac{\{(\text{RRCu})_{\text{FLAT,SURV}}\}_{\text{ADJOINT}}}{\{(\text{RRCu})_{\text{PIN-BY-PIN,SURV}}\}_{\text{ADJOINT}} \cdot \{(\text{RR}^d)_{\text{PIN-BY-PIN,CAVITY}}\}_{\text{FORWARD}}}$$

[†]Similar to above with RR^{NP} replacing RRCu .

*Read 6.18×10^{-17} reactions per second per atom, etc.

Combining Eqs. (2-6) with Eqs. (13) and (16) and decaying the resultant saturated activity to the end of irradiation yields the following expression for the activity when the power spatial distribution is a function of time:

$$A^d(T) = e^{-\lambda_d T} \left(\frac{\text{RR}^d}{\text{RR}^d} \right) \cdot \sum_{j=1}^J C_j \sum_{g=1}^G x_{g,j} \int_{\text{octant}} P_j(r) \phi_{g,d,R}^*(r) dr \sum_{i=1}^{I(j)} F_i [1 - e^{-\lambda_d (t_{i+1} - t_i)}] e^{\lambda_d t_{i+1}}, \quad (17)$$

where $A^d(T)$ = Activity of dosimeter d in dps/atom at end of irradiation,

λ_d = Decay constant of dosimeter d in reciprocal days

T = Irradiation period in days,

RR^d = Saturated activity of dosimeter d calculated in the forward mode using the fluence rate synthesis procedure expressed by Eq. (3) with some reference source,

RR^d = Saturated activity of dosimeter d calculated in the adjoint mode in R - θ geometry using the same reference source as above,

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- J = Number of distinct relative spatial distributions used over T ,
 C_j = Conversion factor relating neutron source and assembly power in units of neutrons/sec per MW,
 $X_{g,j}$ = Relative number of neutrons in group g produced per fission event, normalized to unity,
 $\phi_{g,d,R}^*$ = Adjoint flux in group g (i.e., $g=1$ corresponds to lowest energy group) calculated in an adjoint octally reflected $R-\theta$ run using the response of dosimeter d , in disintegrations per atom per neutron per cm^3 ,
 P_j = The x,y power density distribution in the peripheral core integrated over z based on SPND data and supplemented by diffusion theory pin-by-pin calculations, in MW per cm^3 ,
 $I(j)$ = Number of subintervals describing "instantaneous" total power history for each SPND interval j ,
 F_i = Fraction of total SPND power in fine time interval i due to "instantaneous" power level changes,
 t_i, t_{i+1} = Beginning and end of time subinterval i relative to beginning of irradiation, in days.

Although they will not be detailed here, similar considerations based on adjoint functions for the various group fluence rates in the pressure vessel can lead to expressions for the group fluences arising from time dependent spatial source distributions. In agreement with conclusions deduced from the behavior of the various reaction rates, the group fluence rates were found to scale with predictable accuracies ranging from two to five percent when the scaling takes into account, as it should, the dependence of the group fluence rates on the source fission spectrum as well (10).

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