CONF-900814--1

DE90 009855

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Paper to be given at the Seventh ASTM-UVRATOM
Symposium on Reactor Dosimetry
Strasbourg, France

August 27-31, 1990

*Research sponsored by Nuclear Regulatory Commission, U.S. Department of Energy under Contract No. DE-AC05-84OR21400 with Martin Martieta Energy Systems, Inc.

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The LEPRICON code system is now available from the Radiation Shielding Information Center at ORNL as PSR-277. The system consists of modules that involve both the calculation of neutron fluence rates through PWR pressure vessels and the adjustment of these fluence rates with reduced uncertainties based on surveillance dosimetry. This paper describes in detail the manner in which important parameter uncertainties are partitioned and quantified as part of the input to the adjustment procedure, and to what degree they are applicable to all reactor designs.

INTRODUCTION

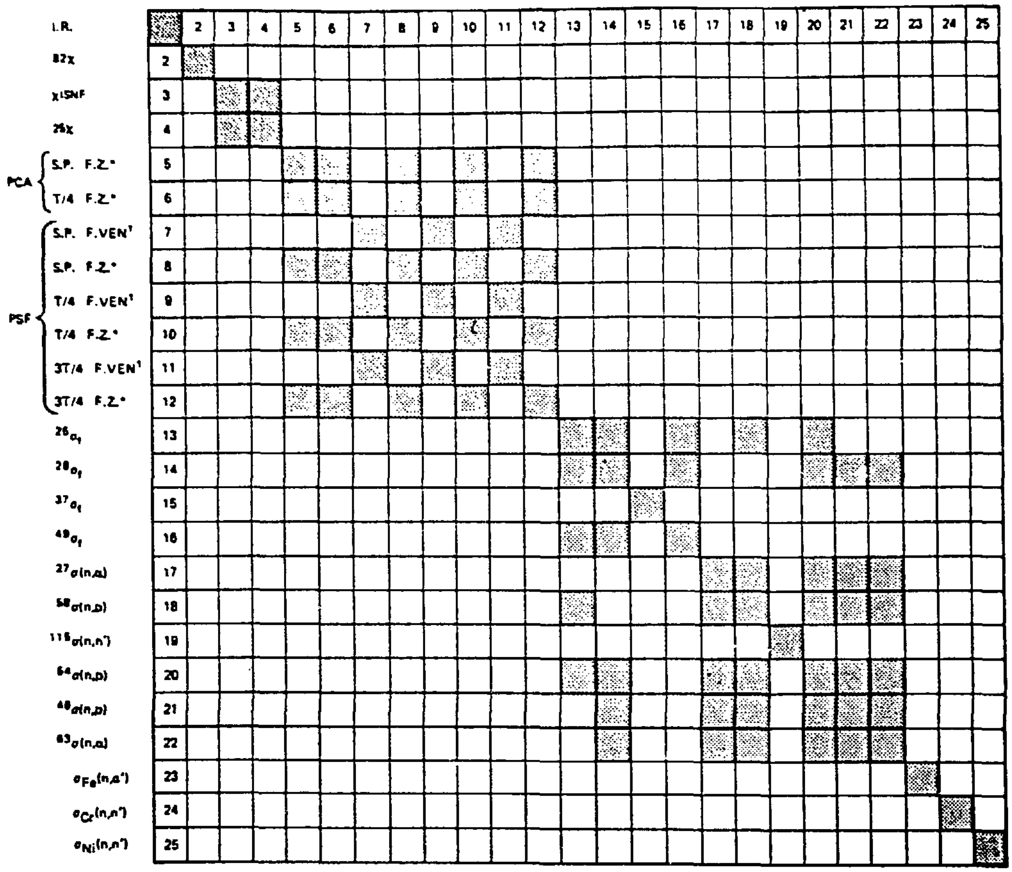
A derivation of the solution to the generalized linear least-squares equations to obtain the best estimate of the group fluence rates in the pressure vessel of a power reactor and their uncertainties is given elsewhere.(1) It consists essentially of two steps. First, the simultaneous combination of all the measured and calculated reactor dosimeter activities with similar benchmark data leads to adjustments in the calculational input data which minimize the discrepancies between all the calculated and measured activities subject to estimates of all the uncertainties involved. Second, adjustments in the calculated pressure vessel group fluence rates (i.e., "unfolding") are determined from the adjustments in the differential data to which they are sensitive. The uncertainties in these adjusted group fluence rates are in general reduced in magnitude from their corresponding values before the adjustment. These best estimates of the group fluence rates are then converted to accumulated group fluences.

THE BENCHMARK DATABASE

The foundation of the LEPRICON system is the "built-in" benchmark database. This database consists of the results of 37 integral measurements in the Intermediate Standard Neutron Field (ISNF), ^{252}Cf, Pool Critical Assembly (PCA), and the Poolside Facility (PSF) fields, in which detailed analyses using the same calculational method and input nuclear data as in the power reactor analysis have previously been performed. A necessary adjunct to this database is a benchmark covariance library which provides values of the estimated group-wise uncertainties of those parameters found to be the most significant in the analyses. These parameters consist of the following nuclear data: the fission spectra from ^{252}Cf, ^{235}U, and a modified ^{235}U spectrum in the ISNF, ten dosimetry cross sections for those reactions analyzed, and the total inelastic cross sections for iron, chromium, and nickel. The remaining parameters are non-nuclear in origin and assume the form of group-dependent flux bias factors. These bias factors provide a means for correcting the calculations for known deficiencies or approximations in the transport method employed (DOT-IV), and uncertainties in both the absolute source strength and its spatial distribution. Two types of bias factors are incorporated into this library: the first, applicable to both the PCA and PSF experiments, represents a correction for the 3-D effects of finite source height; the second, applicable to the PSF only, represents a correction to the absolute magnitude of the leakage from the Oak Ridge Research Reactor determined solely from a VENTURE calculation. The symmetric covariance matrix is shown in Fig. 1, where each shaded square represents the existence of a 15x15 submatrix corresponding to a 15-energy group structure above 0.11 MeV, and which is a subset to the one used in the transport analysis. Blanks indicate that the submatrices are not included in the library. (The first partition, denoted IR in Fig. 1, refers to the covariances of the measured integral responses, and is a 37x37 submatrix.)

EXTENSION TO POWER REACTORS

The extension of the benchmark covariance matrix in Fig. 1 to include additional data for a power reactor first requires an expansion of the integral response submatrix to contain uncertainties in the dosimetry measurements for the reactor. More bias factors must also be appended to the list of parameters in order to introduce a number of possibly significant uncertainties present in the reactor calculations because of the lack of information on the "as-built" physical properties. The bias factors in LEPRICON, whether they apply to the benchmarks or the power reactor, usually have a value of unity, as will be described below. Their uncertainties, however, influence the adjustment procedure, and the adjusted values of the bias factors are no longer unity.



*F.Z. – finite source height bias factor
†F.VEN – VENTURE source bias factor

Fig. 1. Partition extended benchmark covariance matrix

Description of the Bias Factors

Eight types of bias factors applicable to a power reactor are considered in LEPRICON. The first arises as a result of the uncertainty in the exact location of the dosimetry. Although the most probable location (i.e., according to the specifications) is used in the transport calculation, leading to a bias factor of unity, there is an estimated uncertainty in the as-built location of ±1.28cm (see Ref. 2), leading to the considerable calculated group fluence rate uncertainties at the in-vessel surveillance location shown in the column headed "Bias 1" in Table 1. Note that this bias is only valid for a downcomer surveillance position (SP) since any cavity dosimetry (CAV) is not sensitive to uncertainties in its exact location.

A second type of bias arises as a result of an out-of-roundness of the pressure vessel in the vicinity of the location in the pres-

Table 1. Built-in bias factor fractional standard deviations.

Elow Mev	Gr	Bias 1 (SP)	Bias 2 (T/4)	Bias 2 (CAV)	Bias 3 (SP)	Bias 3 (T/4)	Bias 3 (CAV)	Bias 4 (SP)	Bias 4 (T/4)	Bias 4 (CAV)	Bias 5 (SP)	Bias 5 (T/4)	Bias 5 (CAV)	Bias 6 (SP)	Bias 6 (T/4)	Bias 6 (CAV)	Bias 7 (CAV)	Bias 8 (CAV)
11.1	1	0.109	0.068	0.067	0.031	0.037	0.037	0.041	0.041	0.029	0.020	0.020	0.040	0.030	0.030	0.055	0.001	0.050
8.2	2	0.121	0.074	0.073	0.034	0.041	0.040	0.042	0.042	0.030	0.020	0.020	0.040	0.031	0.031	0.058	0.004	0.050
6.1	3	0.133	0.079	0.078	0.038	0.045	0.044	0.042	0.042	0.031	0.020	0.020	0.040	0.034	0.034	0.063	0.008	0.050
4.1	4	0.147	0.088	0.088	0.042	0.050	0.049	0.045	0.045	0.031	0.020	0.020	0.040	0.036	0.036	0.066	0.017	0.050
3.0	5	0.170	0.095	0.092	0.050	0.057	0.055	0.046	0.048	0.031	0.020	0.020	0.040	0.035	0.035	0.064	0.018	0.050
2.6	6	0.178	0.100	0.098	0.053	0.060	0.057	0.048	0.048	0.031	0.020	0.020	0.040	0.033	0.033	0.062	0.015	0.050
2.1	7	0.160	0.102	0.099	0.057	0.063	0.060	0.046	0.046	0.031	0.020	0.020	0.040	0.032	0.032	0.060	0.020	0.050
1.8	8	0.183	0.103	0.099	0.060	0.063	0.060	0.046	0.048	0.031	0.020	0.020	0.040	0.031	0.031	0.058	0.041	0.050
1.5	9	0.193	0.105	0.101	0.063	0.065	0.062	0.046	0.046	0.031	0.020	0.020	0.040	0.030	0.030	0.055	0.037	0.050
1.2	10	0.195	0.106	0.102	0.065	0.065	0.063	0.042	0.042	0.031	0.020	0.020	0.040	0.029	0.029	0.052	0.039	0.050
0.91	11	0.203	0.108	0.103	0.069	0.066	0.065	0.046	0.048	0.031	0.020	0.020	0.040	0.026	0.026	0.045	0.049	0.050
0.61	12	0.205	0.109	0.105	0.072	0.065	0.064	0.048	0.048	0.031	0.020	0.020	0.040	0.024	0.024	0.038	0.054	0.050
0.37	13	0.202	0.109	0.106	0.078	0.070	0.069	0.048	0.046	0.031	0.020	0.020	0.040	0.021	0.021	0.034	0.115	0.050
0.21	14	0.198	0.109	0.107	0.070	0.070	0.069	0.048	0.048	0.031	0.020	0.020	0.040	0.019	0.019	0.028	0.112	0.050
0.11	15	0.184	0.109	0.107	0.075	0.072	0.071	0.048	0.046	0.031	0.020	0.020	0.040	0.019	0.019	0.028	0.156	0.050

sure vessel where the fluences are desired. For pressure vessels fabricated by Combustion Engineering, a value of ±1cm (Ref. 2) has been estimated as the variability in the as-built pressure vessel inner radius, leading to the fluence rate uncertainties shown under "Bias 2" in Table 1. Note that this bias is assumed to affect only the pressure vessel (T/4) fluence rate and any cavity dosimetry. By using the nominal value for the inner radius in the calculations, this bias factor is initially unity.

A third type of bias factor allows for uncertainties in the coolant density in the downcomer region. Assigning a value of ±2% to the uncertainty in this density due to variations in thermal hydraulic conditions during operation leads to the next entries in Table 1, where effects are indicated for all three fluence rate locations. Again these factors are initially unity if nominal axially dependent conditions of temperature and pressure are used in the RZ calculations. If only axially averaged conditions are used in a single RTHETA calculation, initial values may differ from unity and the uncertainties should be increased.

A fourth type of bias factor arises as a result of uncertainties in the radial source distributions throughout the peripheral assemblies (axial distributions are adequately described by in-core instrumentation). In LEPRICON this is assumed to be a calculated pinwise distribution normalized to in-core relative measurements of assembly powers. The group fluence rate uncertainties at all three locations arising from uncertainties of 5% in the calculated relative subassembly distributions (3) and another 5% in the relative measurements (4) are given in the next column in Table 1. The uncertainty in the absolute source normalization is included as an uncertainty in the measured activities, which in LEPRICON are expressed in units of disintegrations/s per target atom per core neutron/s averaged over the dosimeter exposure.

A fifth type of bias factor is included whose uncertainty describes approximations made in the 3-D fluence rate synthesis procedure used in the calculation of the fluence rates, and is identical to the one included in the PCA and PSF analysis to take into account effects of finite source height, which involves both an RZ and R calculation in addition to the RTHETA. The values appearing in Table 1 are estimates of the uncertainties in these unit bias factors from comparisons with Monte Carlo calculations (5).

A sixth type of bias factor is included to account for any variations from nominal values in the densities of steel in the core barrel, thermal shield, and pressure vessel. For pressure vessels fabricated by Babcock and Wilcox (B&W), this bias also accounts for uncertainties of ±0.5cm in the thickness of the vessel (Ref 6). An assumed uncertainty of 1% in the steel density gives rise to the fluence rate uncertainties shown in Table 1. Corrections to the cavity values for Bias 6 in this table for B&W pressure vessels increase these entries by about 70%.

The last two types affect the cavity dosimetry only. Bias type seven arises from uncertainties in the composition of the concrete wall backing the reactor cavity–in particular the water content. The presence or absence of steel linear on the inner surface of this concrete or rebar in the concrete also affects the reflected fluence rates. Reasonable variations in these variables produce the bias uncertainties given in Table 1. Bias type eight is used to correct the calculations for streaming effects in the cavity. It may have non-unit initial values for off-belt-line dosimetry locations in thin cavities. It was not found necessary to activate this bias in any of the previous studies supporting the LEPRICON development and thus little guidance can be offered in its application.

All bias factor types, the two for the benchmarks and up to eight for the reactor, are assumed independent of one another. An energy correlation within each type exists but it is secondary in importance to the group variances, and these autocorrelations can be treated rather cavalierly. In LEPRICON, these correlations within a given bias factor type at a specific location (i.e., a column in Table 1) are assumed to be group independent and high since for the most part the entries in the table were obtained from completely correlated one-dimensional calculations. (The use of unit correlations should be avoided in order to introduce some flexibility into the adjustments, however, since a strictly rigid covariance matrix has some undesirable properties which are not completely realistic).

For bias factor types that are functions of position as well as energy (i.e., types 2-6), cross-correlations are also assumed to exist between fluxes in the same group at the different locations, and these are also assigned high values independent of energy. Thus, to within the degree of detail warranted, the complete correlation between a group fluence rate at one location and another group fluence rate at another location has been approximated as the product of the cross-correlation and autocorrelation coefficients.

Thus far the only power reactors included in the supporting analysis of the LEPRICON system have been of B&W design (AND-1) and Westinghouse design (HBR-2). However, the coding is sufficiently general to allow additional reactor designs to be included. This generality must not be misconstrued to imply that other designs are automatically valid using the current set of defaults shown in Table 1 and without the possible need to introduce new types of bias factors. Additional work might have to be performed outside of the LEPRICON system to arrive at credible estimates of uncertainties for a new type of bias factor peculiar to that reactor. The generality in the code is such that when these new bias factor uncertainties are determined the code will allow them to be substituted as part of the standard input. This redefinition and/or overriding of the default uncertainties of the bias factors combined with the parameterization of two important sensitivities to be described next allows virtually any reactor to be treated by LEPRICON.

SENSITIVITIES AND FLUX UNCERTAINTIES

Propagation of the uncertainties in the calculated fluence rates are determined in a straightforward manner from the uncertainties in the input differential data. They are obtained as quadratic products of the data covariances and the flux sensitivities, where the sensitivities are defined as fractional changes in the group fluence rates per unit fractional change in each group datum:

$$S_\phi = \frac{\partial\phi/\phi}{\partial p/p} \quad , \tag{1}$$

and

$$C_{\phi\phi} = S_\phi C_{pp} S_\phi^T \quad . \tag{2}$$

In Eq. (1) ϕ has dimensions Gx1 and p has dimensions PGx1, where G is the number of groups and P is the total number of parameter types, including both nuclear and non-nuclear data. In Eq. (2) $C_{\phi\phi}$ is the fluence rate fractional covariance matrix propagated from the total parameter covariance C_{pp}, also expressed in fractional values, by the sensitivities S_ϕ ("T" represents the transpose). S_ϕ has dimensions GxPG ("sensitivity profiles"), C_{pp} has dimensions PGxPG, and $C_{\phi\phi}$ has dimensions GxG. For the bias factors, since they represent multiplicative correction factors to the calculated fluence rates,

$$\phi_{g_c} = \phi_{gj} \prod_j (BF_j)_g \quad , \tag{3}$$

and the sensitivities are either zero or unity.

The nuclear data used in a power reactor transport calculation that result in the greatest ex-core fluence rate uncertainties are $^{25}\chi$, the spectrum from thermal fission of ^{235}U, and σ_{STEEL} (n,n'), the total inelastic "steel" cross section. In order that they may be used with any type of reactor design, the sensitivities of the group fluence rates in a power reactor geometry to these two types of differential data have been approximated by empirical expressions. From analysis of rigorous sensitivity calculations for the PCA and PSF benchmarks as well as the experiments in AND-1, the flux sensitivities to σ_{STEEL} (n,n') may be represented to acceptable accuracy as

$$S_\phi = K_\phi t_{STEEL} \quad , \tag{4}$$

where K_ϕ, the sensitivity of group fluence rate g to the total inelastic steel cross section in group k per cm of steel penetrated, is represented by a spatially invariant GxG matrix. This approximation implies the effect of any intervening coolant is small and the effect to be linear in the intervening steel thickness. The latter effect indicates that uncertainties in the cross sections are propagated to all ex-core flux locations. The resulting fluence rate covariances thus vary with position in the reactor, increasing with increasing thickness. A very important consequence of this is that these cross sections affect significantly the overall correlation between the fluence rates at any two locations—e.g., dosimetry and in the pressure vessel— thus causing the parameter adjustments determined at the dosimetry location to result in fluence rate adjustments of different amounts at the two locations. This readily follows from the expression for the covariances between the fluxes at the two locations:

$$C_{\phi_A \cdot \phi_B} = S_{\phi_A} C_{pp} S^T_{\phi B} \quad , \tag{5}$$

in which $S_{\phi A}$ and $S_{\phi B}$ represent sensitivity profiles to p at locations A and B respectively.

A similar analysis of sensitivities to changes in the fission spectrum was performed and led to an even simpler expression than Eq. (4):

$$S_\phi = K \quad , \tag{6}$$

which implies that for a wide range of ex-core locations the sensitivities rapidly approach asymptotic values independent of the specific nature of any intervening material media or the distance from the core. These "universal" sensitivities are represented by another invariant matrix, this time leading to virtually complete correlations to this parameter over all ex-core fluence rate points enclosed within the concrete shield.

CONCLUDING REMARKS

In concluding, it can be stated that the fluence rate uncertainties at a given pressure vessel location tend to be similar for a given class of reactors since the bias factors and sensitivities to the nuclear data tend to be similar. While the same conclusion is expected to remain generally valid for any location–in particular that of a dosimeter–the use of different in-vessel dosimetry locations as well as the possible introduction of ex-vessel dosimetry will result in different fluence rate uncertainties at these new dosimeter locations, but these should not affect to any significant degree the resulting reduced flux uncertainties and adjustments at locations within the pressure vessel.(7)

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