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VALIDATION OF NEUTRON-TRANSPORT CALCULATIONS IN BENCHMARK FACILITIES FOR IMPROVED DAMAGE-FLUENCE PREDICTIONS*

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INTRODUCTION

An accurate determination of damage fluence accumulated by reactor pressure vessels (RPV) as a function of time is essential in order to evaluate the vessel integrity for both pressurized thermal shock (PTS) transients[1] and end-of-life considerations. The desired accuracy for neutron exposure parameters such as displacements per atom or fluence ($E > 1$ MeV) is of the order of 20-30%.[2] However, these types of accuracies can only be obtained realistically by validation of nuclear data and calculational methods in benchmark facilities.[3] The purposes of this paper are

1. to review the needs and requirements for benchmark experiments,

2. to discuss the status of current benchmark experiments and to summarize results and conclusions obtained so far, and

3. to suggest areas where further benchmarking is needed.

A variety of data and calculational procedures enter into the final determination of damage fluence and the operating life of reactor pressure vessels. Different types of nuclear data and various calculational methods need to be validated separately so that inaccuracies introduced by use of one particular data set or method approximation are not obscured by uncertainties in other data sets or procedures. Thus, the complete validation of data and methodology necessary for damage assessment in RPVs requires a chain of benchmark experiments of increasing complexity such that the simpler experiments adequately test the basic nuclear data while the more complicated ones test important cross sections employed in the

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transport process, the method approximations used in the transport calculation, and the geometric approximations used in modeling the experiments. The gamut of benchmarks consequently should run from dosimetry measurements performed in a "point" ^{252}Cf field to accurate surveillance measurements performed during an adequately documented cycle in an operating power reactor.

The Nuclear Regulatory Commission (NRC) has sponsored benchmark experiments in different test and power reactors in the U.S. and is cooperating with the Electric Power Research Institute (EPRI) and several European countries to validate RPV damage analysis methods. These benchmark experiments actually can serve a dual purpose. First, they provide a verification of calculational procedures and nuclear data used by a particular organization, as described earlier. Second, NRC has sponsored "blind test" calculations[4] to determine the spread of computed results to establish a measure of the calculational uncertainty in RPV calculations. Table 1 summarizes these benchmark experiments and their particular purpose in the overall validation program.

VALIDATION OF ANALYTIC METHODS

Validation of procedures and data through benchmark experiments is usually done by comparing discrete ordinates transport calculational predictions with certified experimental results. Least squares adjustment procedures can then be used to correct nuclear data and to reduce uncertainties in calculational results. Such adjustment procedures are also recommended in the determination of damage fluence in RPVs using surveillance capsule dosimetry measurements. A variety of suitable adjustment methods is listed in ASTM Standard E944. EPRI has sponsored the development of the LEPRICON code system (Fig. 1) at ORNL in an attempt to standardize the codes used by utilities for RPV fluence analysis.[5] The source data for this code system were obtained from the NRC-sponsored benchmark experiments.

There are three major sources of uncertainty in discrete ordinate transport calculations of RPV fluence. These are

1. uncertainties in the nuclear data,
2. geometric approximations in modeling the reactor configuration, and
3. numerical approximations.

The most important nuclear data for RPV analysis that need validation are the following:

1. iron total inelastic scattering cross section,
2. dosimeter cross sections (needed to compute dosimeter reaction rate for comparison with calculations), and
3. ^{235}U fission spectrum.

The most common modeling approximations and uncertainties which need validating are the following:

1. calculation of the core fission source using reactor core physics methods,

2. interpolation of X-Y source to an R-θ grid,

3. synthesis of one- and two-dimensional discrete ordinates calculations to produce three-dimensional fluxes,

4. effect of space-time variation of the reactor power,

5. effect of perturbation of the surveillance capsule, and

6. effect of cavity streaming on ex-vessel dosimeter calculations.

Numerical approximations such as selection of the proper quadrature, mesh spacing, etc. are assumed to be treated adequately by knowledgeable personnel who perform the transport calculations and will not be addressed here. However, the effects of data uncertainty and modeling approximations are difficult to ascertain, and benchmark configurations are required to validate their adequacy.

It has been demonstrated in the PCA benchmark experiment that the uncertainty in calculated RPV fluxes may be reduced by about a factor of two by using the LEPRICON adjustment module. However, it would be a mistake to believe that any least squares adjustment program can correct for grossly or even moderately inaccurate transport calculations; and those analyses which rely upon use of experimental data to correct a basically flawed transport calculation will be in error.

Independent validation of the calculational procedures in benchmarks is, therefore, indispensible. In particular, the determination of damage fluence relies on accurate calculations for the following reasons:

1. Least squares adjustment codes require that the computed results ("prior data") be close to the true results, in order that the first order perturbation assumption be valid.

2. In RPV damage analysis, the locations in the RPV that are of interest are often far removed from the location of the surveillance dosimeters so that calculations must be heavily relied upon for extrapolation.

3. Surveillance dosimeter results are only available at a few times during the lifetime of the plant.

4. Studies examining alternate core or future design configurations do not have access to dosimeter results.

RESULTS OF BENCHMARK EXPERIMENTS

With the exception of two important areas (to be discussed shortly), the current benchmarking programs are sufficient to validate most of the transport methods and data needed for RPV analysis. Several of the benchmark experiments have been completed, while others are still being performed. The preliminary results that are being obtained from these benchmarks indicate that acceptable accuracy (better than 20%) can be obtained for midplane calculations in which the core source is well characterized. The investigations to date lead to the following preliminary conclusions:

1. It has been shown that the ENDF/B-V dosimetry cross sections can be consistently adjusted to produce better agreement among many differential experiments.[6] The major effect of one such modification is a reduction of about 9% in the $^{63}Cu(n,\alpha)$ cross section, which also leads to better consistency in the integral results in the ORNL Poolside Facility (PSF) benchmark analysis.

2. Midplane calculations of dosimeters in simulated RPV configurations, such as mocked up in the Pool Critical Assembly (PCA) and the PSF, can be computed to an accuracy of 10-20% using discrete ordinates transport theory. By using the LEPRICON least squares adjustment procedures* to improve the calculations, the uncertainty of adjusted fluxes at the midplane T/4 position in a simulated vessel can be reduced by almost a factor of two by adding the information from surveillance measurements and other benchmark data.

3. The PCA benchmark experiments[5] indicate that the iron total inelastic cross section in ENDF/B-V should be reduced by about one standard deviation (6%) above 3 MeV, and the ^{235}U fission spectrum increased by about 0.7 of a standard deviation (10%) above 6 MeV.

4. In the PSF Westinghouse Perturbation Experiment,[9] it was found that the effect of the surveillance capsule perturbation is well predicted by including a representation of the capsule in the transport calculation. For Westinghouse capsules, the perturbation to dosimeter reaction rates was found to be up to 33% for a ^{237}Np dosimeter.

5. Comparisons of calculations with measured fission rates in the VENUS benchmark reactor core indicate the shape of the neutron source can be computed to an accuracy of about 5% near the important core-baffle interface using transport theory.[10,11] However, no validation of diffusion theory calculations has been done.

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*So far only LEPRICON has been actually applied to the simultaneous adjustment of RPV fluences from surveillance measurements and other benchmark data. However, the potential exists in a few other adjustment codes.[7,8]

6. The LEPRICON 3-D flux synthesis method[9] has been benchmarked for the PCA and the ORR-PSF, two small research reactors. This approximation was able to accurately predict axial and horizontal reaction profiles within 120 mm of the midplane, even with asymmetries in the core source. This distance corresponds to about 40% of the distance to the top of the active ORR core. However, the synthesis method has not been validated near the top of the active core, nor for large reactor cores, nor for cavity calculations in which streaming may affect the accuracy.

7. Cavity measurements and calculations in the 2568 MW_{th} ANO-1 PWR indicate that ex-vessel dosimeters at the midplane elevation can be computed to an accuracy of about 10%, but the agreement at an elevation corresponding to the coolant inlet nozzle is only within about 30%.* It is not presently known if this discrepancy is caused by difficulty in computing cavity streaming or by a breakdown of the 3-D synthesis method or by some problem with the dosimeter measurements.

NEED FOR FUTURE BENCHMARKS

The present benchmark program has done much toward validating the methods and data used in RPV damage fluence calculations; however, there are two areas in which it is felt that further benchmarking could significantly enhance the validation program. The first of these is an extension of the present VENUS benchmark to include validation of "standard" core analysis methods such as used by utilities. While the present VENUS program is establishing that the power density of the core periphery can be accurately computed with transport theory, it is important to validate the accuracy of two-group diffusion theory at these locations, since that is the method used for most PWR core analysis and which may be employed in providing a relative pin-by-pin power spatial distribution for subsequent transport calculations of ex-core fluxes. It would also be desirable to establish the validity of diffusion theory for computing the power distribution near "shield assemblies," which have been suggested as a means to reduce RPV fluence.

The most crucial calculational method still lacking validation in a commercial power reactor, however, is the LEPRICON (or some other) 3-D flux synthesis approximation in the RPV above the top and below the bottom of a PWR core. These locations are often the most critical for PTS damage analysis, and yet it has not been validated that the synthesis procedure will produce reliable results in such extreme axial positions. A benchmark experiment which simulates the upper or lower core region, the in-vessel geometry, and the RPV, and which provides axial traverse measurements would be very beneficial to the methods verification program.

*Unpublished results from R. E. Maerker and M. L. Williams.

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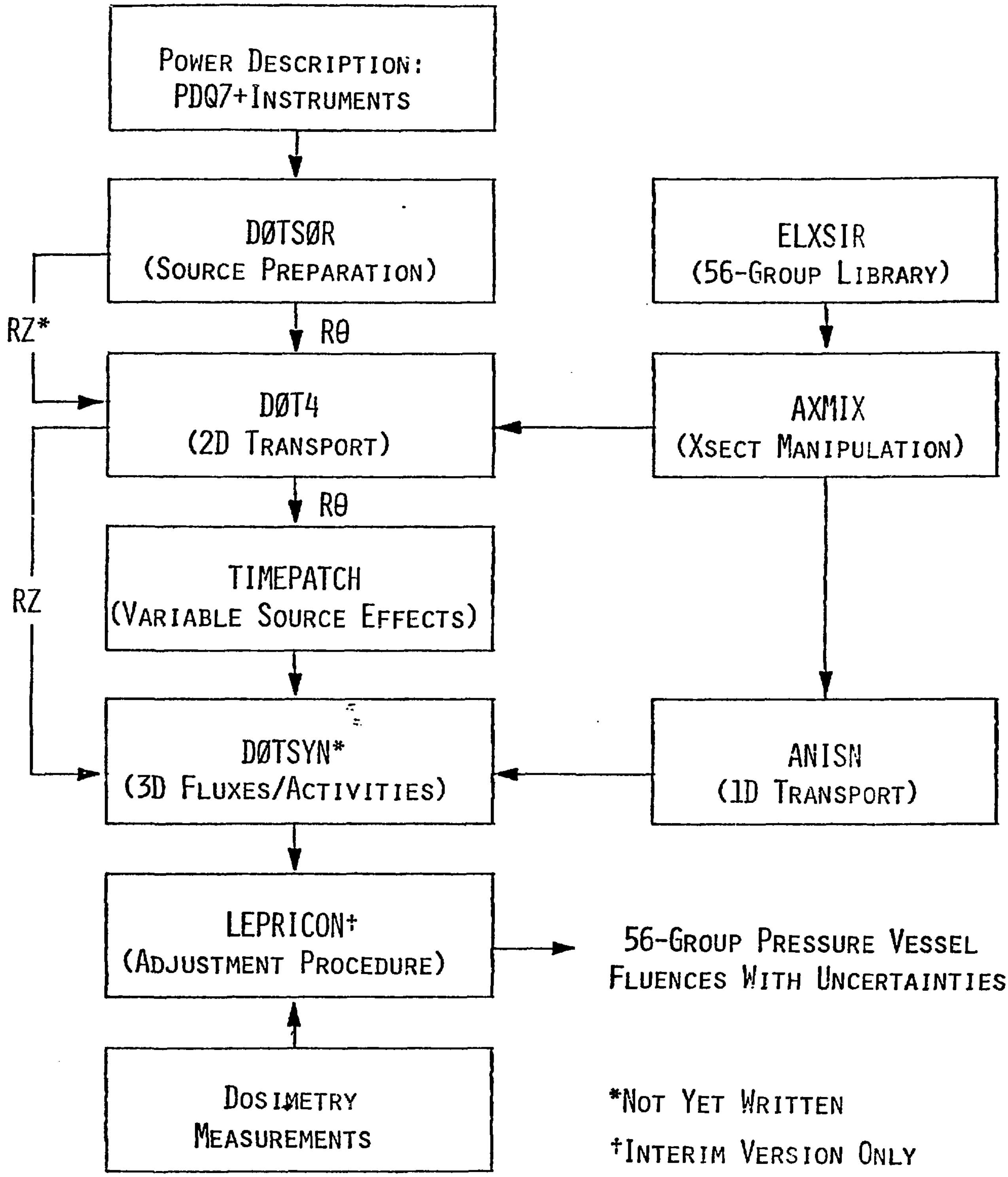

FIG. 1. LEPRICON SYSTEM FLOW CHART.

Table 1. Benchmark experiments

Facility	Location	Purpose in overall validation program*	Status
^{252}Cf	NBS	Tests integral fission cross sections averaged over a spectrum above approximately 0.1 MeV.	Completed
^{252}Cf	PTB	Tests integral reaction rate cross sections averaged over a spectrum above approximately 0.1 MeV.	Completed
ISNF	NBS	Tests integral fission cross section ratios averaged over a moderated ^{235}U fission spectrum. Also tests the ^{235}U spectrum.	Completed
PCA	ORNL	Tests how accurately transport calculational methods can predict neutron and gamma fluxes in a simulated 2-loop PWR steel water configuration. Tests diffusion and transport theory codes to calculate the fission source distribution. Tests iron cross sections, effects of a finite source (i.e., flux synthesis approximation), and sensitivity to ^{235}U fission spectrum. Establishes a measure of the transport calculational method uncertainty by means of a "blind test."	Completed
PSF	ORNL	Tests 3-dimensional (3-D) diffusion theory fission source generation and transport calculational methods (as specified under PCA) that takes into account the power time history of an operating reactor. The primary purpose of the PSF was to establish a measure of the uncertainty involved in the surveillance procedures used by vendors and service laboratories to predict damage in the pressure vessel from data taken at an accelerated surveillance location. Provides measurements at off-axis positions both inside and outside the PV to validate calculational results at locations important to the PTS problem (tests flux synthesis approximation).	Continuing

Table 1. (continued)

Facility	Location	Purpose in overall validation program	Status
SDMF	ORNL	To provide certification of vendor and service laboratories capabilities to calculate the flux perturbation caused by the physical presence of a surveillance capsule. Provides experimental verification of how well RM, SSTR, HAFM, and DM sensor results can be correlated with the calculated PV steel dpa gradient from a surveillance capsule to and through the PV.	Continuing
VENUS	CEN/SCK	To validate calculational estimates of core source distributions, on a pin-to-pin basis for the last fuel row, in terms of total absolute core power (LWR core management for mitigation of PTS). To validate calculational estimations of core boundary heterogeneity effects, and, in more recent plants, of the heterogeneity effect of neutron pads attached to the core barrel (azimuthal effects). To validate gamma heating calculations in LWR core internals.	Continuing
NESDIP	AEEW	Provide a "cavity benchmark" against which procedures for correlating cavity measurements with exposure parameters inside the PV can be tested. Provide measurements of the effects of narrow and wide reactor cavities on ex-vessel dosimetry.	Continuing
ANO-1	AP&L	Tests geometry modeling of a typical B&W reactor, including effects of axial streaming in the reactor cavity. Also tests accuracy of method used to generate the fission source.	Continuing
ANO-2	AP&L	Tests geometry modeling of a typical CE reactor, including effects on in-vessel and ex-vessel measurements.	Continuing

*The overall validation program encompasses more items than the validation of the transport calculations covered in this paper.