UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REACTOR PRESSURE VESSEL FLUENCE METHODOLOGY

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNITS 1 AND 2

NORTH ANNA POWER STATION, UNITS 1 AND 2

1.0 INTRODUCTION

By letter dated June 18, 1998, Virginia Electric and Power Company (VEPCO/licensee), submitted information and requested review and approval of its proposed pressure vessel fluence methodology described in the topical report VEP-NAF-3 "Reactor Vessel Fluence Analysis Topical Report" (Reference 1).

The proposed methodology has been benchmarked against the results of: (1) the PCA experiment (Configuration 12/13), (2) the results of dosimetry from 10 surveillance capsules from the Surry and North Anna plants and (3) ex-vessel dosimetry from the Surry Cycle 13. The experimental results were statistically analyzed and a few measurements outside acceptable limits were rejected. The rejection was based on statistical arguments.

2.0 EVALUATION

2.1 Methodology

The methodology employs two programs: the DOT code based on discrete ordinates (Reference 2) and MCNP, a Monte Carlo code (Reference 3). The DOT code is used to determine neutron fluxes in the (r,θ) and (r,z) dimensions and synthesize the fluxes for three dimensionional solutions. The flux synthesis is based on the equation $\varphi(r,\theta,z,E)= \varphi(r,\theta,E)*\varphi(r,z,E)/\varphi(r,E)$. The MCNP code is used to calculate parts of the core without axial continuity of the fuel and of the corresponding neutron source. This is the case for some peripheral assemblies where part of the fuel is hafnium-sleeve covered to suppress the flux locally for the protection of vessel welds. DOT is a multigroup energy model while the MCNP is a continuous energy three dimensional model. A number of auxiliary routines are used to prepare the input, convert geometric parameters, plot input and/or output, etc. For the estimation of the projected end-of-life (EOL) fluence, a 90 percent load factor is assumed based on the historical performance of the Surry and North Anna plants. An octal symmetry is assumed for the geometrical set-up of the DOT problem. The azimuthal intervals are less than 1° with 137 radial intervals. (The bootstrapping technique is not used.) The fuel (source) areas and the peripheral assemblies are represented to an accuracy of better than 0.01 percent. The mesh spacings are adjusted to represent the upper and lower reflector regions and the former plates located between the core barrel and the core baffle.

The water density in the downcomer, the bypass, and the fuel region are represented on a cycle-specific basis. The densities of the vessel steel and the internals steel and stainless steel are based on nominal design values. The fuel is modeled as having a burnup of either 0 MWD/MTU or 45,000 MWD/MTU. The licensee claims that this approximation does not introduce a significant error and that the error is accounted for in the uncertainty evaluation. The MCNP material compositions are the same as those used in the DOT input for consistency.

The 47 group BUGLE-93 transport cross section library is used with the DOT code (Reference 4). The P_3 Legendre expansion is used for the scattering cross section. The BUGLE-93 cross sections have been benchmarked against the 199 energy group VITAMIN-B6 library using the PCA test configuration 4/12 and the results agreed to within 4 percent. The MCNPDAT6 library is used for the MCNP code (Reference 5). These data are for continuous energy and are not using the energy group formulation.

The spatial neutron source distribution is derived from three-dimensional power distributions calculated using the PDQ two zoned model (Reference 6). The cycle-specific source is the average of several burnup steps over the length of the cycle. For the transport calculations, the source distributions, to be suitable for the (r,θ), (r,z) and (r) calculations, are processed with the DOTSOR code (Reference 7). The (r,θ) distribution represents the PDQV2 power distribution integrated over the height of the fuel. The (r,z) and the (r) distributions represent the (r,z) plane at the location of the peak vessel fluence. The neutron source for the MCNP calculations are similarly converted using the MCNPSRC code. The source spectrum is a weighted average of the fission spectra for U-235, U-238, Pu-239 and Pu-241. The weighting factors are cycle-specific and are based on the average burnup and the original enrichment of the peripheral assemblies. In addition to the P_3 scattering cross section approximation, the S_8 angular quadrature is used with a θ-weighted difference flux extrapolation and a point flux convergence criterion of 0.001. The adequacy of all of the above approximations were tested and verified.

While the above approximations are not applicable to MCNP, a bias can be introduced from the finite size of the tallies. This is of particular concern in the area of the peak fluence. Sensitivity studies indicated that a 5° tally (containing the peak) has a 2% bias; thus, the flux of such a tally is increased by 2%. At the azimuthal minimum the bias is in the opposite direction; however, because this coincides with axial welds, the flux is not reduced, and that is conservative. A number of statistical checks are performed to assure that both the tally mean value and the associated uncertainty are acceptable.

The proposed methodology and its application meet staff recommendations and the requirements of the draft RG-1053, and they are acceptable.

2.2 Benchmarking

The vessel fluence methodology was benchmarked using a combination of (1) pressure vessel simulation experiments (PCA), (2) plant-specific surveillance capsule measurements, and (3) Surry 1, cycle 13 cavity dosimetry measurements. Both codes were used for the calculation of the PCA experiment, and in both cases the same approximations and cross sections were used.

The results of the PCA vessel simulation indicated that for E > 1.0 MeV the calculation overestimates the measured fluxes by 4 percent to 10 percent.

Data from 10 of the Surry and North Anna surveillance capsules were also used for benchmarking. The information for the available capsules was obtained from the Westinghouse "Surveillance Capsule Reevaluation Report" (Reference 8). Both the DORT and the MCNP codes were used in the analysis for all of the fuel cycles using cycle-specific power and power distributions. The results of the analysis indicated that the mean calculated/measured (C/M) value was 1.02 with a standard deviation of 12 percent. However, individual dosimeter measured values exceeded the corresponding calculated value by 20 percent. These were the Cu-63 dosimeters from the T, W and V capsules from Surry 1 and the Np-237 dosimeter from the Surry 2 capsule W. No reasonable explanation could be found for the behavior of these dosimeters and they were discarded. The mean C/M value of the remaining dosimeters is 1.02 with a standard deviation of 9.05 percent.

The MCNP code was used to perform a limited number of dosimeter analyses, mainly to show that the MCNP code results agree with those of the DOT code. The Surry 1 capsules were chosen to provide additional analysis for the Cu-63 dosimeters. The MCNP results matched the corresponding DOT values to within a few percent. The Cu-63 results were about the same as with DOT. For capsule W the average percentage deviation of MCNP calculated and measured was -16.8 percent, but that was for only two values.

Finally, ex-vessel dosimetry was used for the benchmarking. Dosimeters were installed in Surry Unit 1 at the azimuthal locations of 0° and 45° through the 13th cycle. Fe-54, Ni-58, Cu-63, Np-237, U-238, wires of Co/Al and stainless steel were exposed, retrieved, and measured. The dosimeters were analyzed using the MCNP code because the Surry 1 Cycle 13 included partial length fuel assemblies with hafnium-sleeve flux suppressors at 0° and 45° which result in an axially asymmetric flux distribution. The important structural features in the 0° to 45° segment were modeled in the three dimensional MCNP calculation. Other input parameters used in the ex-vessel analysis were the same as those used in the analysis of the surveillance capsules. Comparison of measured and calculated values indicated agreement within 20% with Fe and Ni showing the largest deviations. Comparison of the axial Fe and Ni measurements indicates that there may be a small positive bias in the calculated values for both sets of dosimeters. Three measurements were found to have measured values much lower than the nearby dosimeters and no reasonable explanation could be provided; thus, they were rejected. The remaining values are well within 20 percent of the calculated values. We found the above results of the benchmarking reasonable and adequate; therefore, we find the benchmarking acceptable.

2.3 Analytical Uncertainty Estimates

An uncertainty analysis was performed to estimate the expected accuracy of the methodology. Sixteen sources of uncertainty were identified and their contributions were estimated with sensitivity analyses. The largest contributors were identified as (1) Fe inelastic scattering cross sections, (2) vessel out-of-roudness, (3) source distribution and (4) fission spectrum. The identified uncertainties were statistically combined and the total 1σ uncertainty is given below.

- DOT peak fluence locations .... 16.4 percent
- MCNP peak fluence locations ....17.0 percent
- DOT upper circumferential weld ....18.1 percent
- MCNP upper circumferential weld ....17.9 percent
- DOT welds shadowed by hafnium inserts ....16.4 percent
- MCNP welds shadowed by hafnium inserts .17.0 percent

A small bias was identified which was non-conservative at the flux peak location and conservative in the minimum location, and for the non-conservative location a correction factor was added to the model. A larger conservative bias (about 10 percent) was identified, which was caused by the PDQV2 source distribution around the part length hafnium flux suppression inserts. However, this bias has not been removed.

The uncertainties and the biases appear reasonable. The fact that the conservative bias due to the source distribution was not removed is conservative. We find the proposed uncertainties acceptable.

2.4 Compliance with Draft DG-1053

Appendix 1 to the methodology includes a summary of the requirements of the Draft DG-1053 and the corresponding justification that the methodology meets the requirements. We find the justification acceptable.

3.0 SUMMARY, CONCLUSIONS AND LIMITATIONS

The NRC staff has reviewed the "Reactor Vessel Fluence Analysis Methodology" submitted by the Virginia Electric and Power Company, the licensee for the Surry and North Anna nuclear power plants. We found the proposed methodology to be acceptable for referencing in licensing actions. This finding (as indicated in the above review) is based on the fact that the proposed methodology utilizes methods and approximations recommended by the staff. The licensee used three different methods to benchmark and verify the results of the calculations and to estimate the analytic uncertainty. The first method used results from the PCA experiment, which the staff recommends, and it is acceptable. The second method utilized the results of 10 Surry and North Anna surveillance capsules. The number of capsules is not large, but the capsules are considered to be plant-specific because the methodology is intended for the Surry and North Anna plants. In addition, the database used in the benchmarking is free of unexplained biases or deviations and appears to be normal. The licensee used the results of reactor cavity dosimetry as a third means for a benchmark. The results showed larger deviations than the capsule results which is what is expected from cavity measurements. The licensee estimated the analytic uncertainty, and it is within recommended and acceptable limits. The components of the uncertainty include all of the potential major contributors. The licensee performed a comparison of the Draft Guide DG-1053 and the methodology to demonstrate that the proposed method complies with these requirements. We found that the methodology meets the requirements of the draft guide.

The approval of this methodology is subject to the following limitations:

- it is applicable only to the Surry and North Anna plants;

- if the assumed load factor of 0.90 changes and the method is utilized for the estimation of projected fluence values, the value of the load factor will be adjusted accordingly; and
- the licensee will assure compliance with DG-1053 when it is published in its final form.

Principal Contributor: L. Lois

Date: April 13, 1999

REFERENCES

1. Letter from J. P. Hanlon, Virginia Electric and Power Company to USNRC "Virginia Electric and Power Company, Surry and North Anna Power Stations, Units 1 and 2, Surry 1, Reactor Vessel Surveillance Capsule X Analysis Report GL92-01, Rev. 1, Supplement 1, Response to Request for Additional Information, and Topical Report on Reactor Vessel Fluence Analysis Methodology" dated June 18, 1998.

2. Rhoades W. A. and R. L. Childs, "An Updated Version of the DOT4 One- and Two-dimensional neutron/photon Transport Code" ORNL-5851, Oak Ridge National Laboratory, July 1982.

3. Briesmeister J. F. Editor, "MCNP - A General Monte Carlo N-Particle Transport Code Version 4A" LA-12625-M, Los Alamos National Laboratory, November 1993.

4. "BUGLE-93, Production and Testing of the VITAMIN-B6 Fine Group and the BUGLE-93 Broads Group Neutron/Photon Cross Section Libraries Derived from ENDF/B-VI Nuclear Data" RSIC Data Library Collection, DLC-175, Oak Ridge National Laboratory, February 1994.

5. Hendrics J.S. et al., "ENDF/B-VI Data for MCNP", LA-12891, Los Alamos National Laboratory, December 1994.

6. VEP-NAF-1, "The PDQ Two Zone Model" by R.A. Hall, Virginia Power Company, July, 1990.

7. Williams M.L., "DOTSOR: A Module of the LEPRICON Computer System for Representing the Neutron Source Distribution in LWR Cores," EPRI Interim Report, December, 1985.

8. WCAP-14040, "Westinghouse Surveillance Capsule Neutron Fluence Reevaluation," by E. P. Lippincott, Westinghouse Electric Corporation, April 1994.