September 14, 2001

Mr. James F. Klapproth, Manager
Engineering & Technology
GE Nuclear Energy
175 Curtner Ave
San Jose, CA 95125

SUBJECT: SAFETY EVALUATION FOR NEDC-32983P, "GENERAL ELECTRIC METHODOLOGY FOR REACTOR PRESSURE VESSEL FAST NEUTRON FLUX EVALUATION" (TAC NO. MA9891)

Dear Mr. Klapproth:

By letter dated September 1, 2000, GE Nuclear Energy (GENE) submitted the subject licensing topical report (LTR) and requested staff review and approval for boiling water reactor (BWR) licensing actions. Additional information was submitted on December 20, 2000, January 5 and 17, 2001, March 2 and 14, 2001, and June 1 and 15, 2001. The NRC staff and Brookhaven National Laboratory (BNL staff consultant) exchanged information with GENE personnel on several occasions in the course of this review.

The proposed methodology employs an analytic approach based on the discrete ordinates neutron transport method to determine the fast (E > 1.0 MeV) flux (and fluence) in BWR vessels. The proposed methodology adheres to the guidance in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The method is using available BWR surveillance capsule dosimetry measurements for the validation of the analytic transport calculations and the estimation of the uncertainty and bias. In addition, the method is compared to the NUREG-6115 benchmark problem and the results of a foreign reactor benchmark provided by GENE.

The staff finds the proposed methodology acceptable for referencing in licensing actions, subject to the limitation that the applicant will demonstrate the method's predictive capability in at least four surveillance capsules within three years from the day of approval of this methodology. The LTR includes a limited amount of information on the method's capability to predict the fluence on and through the core shroud. The staff concluded that the method would yield a conservative fluence estimate on the shroud. In view of the shroud fluence requirements, the staff finds the method acceptable subject to the limitations listed in the summary and limitations section of the enclosed safety evaluation (SE).

A conference call was held on August 14, 2001 between GENE, BNL and the NRC staff to discuss GENE's findings from their review of the draft SE (ADAMS accession no. ML012410011) regarding proprietary information. The conference call determined that there was no proprietary information contained in the SE. GENE requested clarification on the three year requirement for the confirmatory and predictive dosimetry for the vessel and the shroud. The staff stated that: (1) the measurement to calculation comparisons need only include activation dosimetry, (2) RG 1.190 contains the required guidance, and (3) GENE must prepare and submit to the staff a plan, identifying proposed surveillance capsules and a time schedule to satisfy and erase the limitations from the methodology.

The NRC requests that the GENE publish an accepted version of the revised NEDC-32983P within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract, and add an "-A" (designating accepted) following the report identification number (i.e., NEDC-32983-A).

If the NRC's criteria or regulations change so that its conclusion in this letter that the LTR is acceptable is invalidated, GENE and/or the applicant referencing the LTR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the LTR without revision of the respective documentation.

If you have any questions, please contact Robert Pulsifer, GENE Project Manager, at (301) 415-3016.

Sincerely,

/RA/

Stuart A. Richards, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 710

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GE NUCLEAR ENERGY TOPICAL REPORT NEDC-32983P

"GENERAL ELECTRIC METHODOLOGY FOR REACTOR PRESSURE

VESSEL FAST NEUTRON FLUX EVALUATIONS"

PROJECT NO. 710

1.0 INTRODUCTION

By letter dated September 1, 2000, GE Nuclear Energy (GENE) submitted their methodology for reactor pressure vessel fast neutron flux evaluations and requested NRC review and approval (Reference 1). The proposed methodology is intended for the determination of the fast neutron fluence accumulated by the pressure vessel and internal components of US boiling water reactor (BWR) plants. The methodology has evolved from earlier GENE fluence methods. The proposed licensing topical report (LTR) (NEDC-32983P) fluence evaluation employs an analytic approach using the most recent fluence calculational methods and nuclear data sets. In the proposed methodology, the vessel fluence is determined by a discrete ordinates transport calculation in which the core neutron source is explicitly represented and the neutron flux is propagated from the core through the downcomer and the jet pumps and jet pump risers whenever present, to the vessel (rather than by an extrapolation of the measurements). The method proposed for predicting the dosimeter response and the vessel inner-wall fluence is generally consistent with Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” (Reference 2).

The LTR provides a description of the application of the proposed methodology to the calculation of the Brookhaven National Laboratory (BNL) pressure vessel fluence benchmark problem described in NUREG/CR-6115 (Reference 3). The LTR also describes the application of the methodology to the analysis of a GENE dosimetry benchmark experiment (References 4 and 5). This includes a description of both the discrete ordinates DORT (Reference 6) and the MCNP (Reference 7) Monte Carlo transport calculations of the measurements and the techniques used to interpret the in-vessel dosimeter response. Representative BWR surveillance measurements and comparisons to GENE calculations are provided as additional qualification of the calculational methods. The GENE dosimetry measurements are used to validate the DORT vessel fluence methodology and determine the calculational biases and uncertainties.

The LTR fluence calculation and uncertainty methodology is summarized in Section 2. The evaluation of the important technical issues raised during this review is presented in Section 3 and the summary and limitations given in Section 4. The staff was assisted in this review by BNL personnel as consultants.

2.0 SUMMARY OF THE NEDC-32983P VESSEL FLUENCE METHODOLOGY

2.1 Pressure Vessel Fluence Calculation Methodology

The proposed methodology provides a best-estimate prediction of the fluence rather than the conservative prediction as was the case with earlier methods. The fluence calculations are performed with the DORT discrete ordinates transport code. The LTR provides a description of the DORT calculation used to determine the vessel fluence, as well as the calculations used to predict the GENE measured dosimetry and validate the transport model. The calculational model includes a representation of the peripheral fuel assemblies and the core-internals, downcomer and vessel geometry. Calculations are performed to determine the pin-by-pin and bundle-average power distribution in the peripheral fuel bundles for input to the DORT core neutron source. Calculations employ a relatively fine (r, θ, z) spatial mesh and are carried out using both an S_8 and an S_{12} angular quadrature set.

The eighty-group MATXS (Reference 8) cross section library is the basic nuclear data set. This library is used in performing the energy and spatial self-shielding and removal calculations. The scattering cross sections are represented using a P_3 Legendre expansion. The calculations are performed in (r, θ) and (r, z) geometries. A synthesis technique is used to determine the three-dimensional fluence distribution and to some extent account for the effect of axial leakage between the core and the cavity.

Predictions of the dosimeter response measurements are required to determine the calculation-to-measurement (C/M) data base used to validate the fluence calculation methods. The predictions are made for the in-vessel dosimetry using essentially the same methods used to determine the vessel fluence. The proposed methodology includes dosimeter response adjustments for the half-lives of the reaction products and the core power history. In order to ensure an accurate prediction of the dosimeter response, a detailed spatial representation of the capsule geometry is included in the DORT model. The measured dosimeter reaction rates are calculated using the dosimeter-specific reaction cross sections. The calculated dosimeter response is determined for the irradiation period up to the time the capsule was withdrawn.

2.2 Calculation of the BNL Pressure Vessel Fluence Benchmark Problem

As part of the qualification of the fluence calculational methodology, GENE has calculated the BNL NUREG/CR-6115 BWR pressure vessel fluence benchmark problem. The NUREG/CR-6115 report provides the detailed specification and corresponding numerical solutions for the BWR fluence benchmark problem. The calculation of the benchmark problem allows a detailed assessment and verification of the numerical procedures, code implementation, and the various modeling approximations relative to a representative BWR operating configuration. The geometry, materials and space and energy dependent source are fixed by the problem specification and the reference solutions allow comparisons of the predicted fluence at the vessel locations of interest.

The LTR describes the calculations performed using both the proposed DORT discrete ordinates method and the MCNP Monte Carlo method. The DORT calculations were performed using the proposed “current” method. The calculational model included the complete radial geometry from the core out through the concrete biological shield and axially from the core inlet up to the steam separator. As part of the analysis of the benchmark problem, GENE performed a series of sensitivity calculations in which various modeling assumptions were evaluated. Calculations were performed with three downcomer models: (1) a conservative model in which the jet pumps and risers are neglected, (2) an approximate model in which the materials of the jet pumps and risers are homogenized over the volume of the downcomer, and (3) a model in which the components in the downcomer are treated explicitly as heterogeneous material zones. Calculations were performed using both ENDF/B-V and ENDF/B-VI nuclear data sets. The effect of using a more accurate angular quadrature set was evaluated by comparing calculations performed with S_8 and S_{12} quadratures. The effect of the peripheral radial flux gradient on the core neutron source and vessel fluence was evaluated by calculating the fluence with: (1) a model in which a uniform bundle-average power is assigned to each peripheral fuel bundle, and (2) a model in which spatially dependent power distributions (provided with the problem specification) are assigned to the outer three rows of fuel bundles.

Comparisons of the GENE and BNL NUREG/CR-6115 vessel fluence predictions are provided. As an additional verification of the GENE fluence methodology, GENE has performed MCNP Monte Carlo calculations of the BNL vessel fluence benchmark problem. The MCNP model included an essentially exact octant representation of the core, shroud, jet pump/riser and vessel geometry specified in the NUREG/CR-6115 report. The calculations were performed using a continuous energy representation of the nuclear data. The cross section data used in these calculations is based on the ENDF/B-V nuclear data except for iron, hydrogen and oxygen. Since the cross sections for these elements have changed significantly in the more recent ENDF/B-VI data set, ENDF/B-VI cross sections were used for iron, hydrogen and oxygen. Calculations were performed using two models for describing the power/source distribution in the peripheral fuel bundles: (1) a uniform bundle-average power model, and (2) a pin-wise power distribution model. The source normalization used in the MCNP calculation was taken to be the same as that used in the DORT calculation of the benchmark problem. Variance reduction was accomplished by defining a set of importance regions which allowed particle splitting. In addition to the base calculation, a series of MCNP sensitivity calculations was performed to determine the effect of: (1) including the jet pumps and riser materials, (2) variations in the fuel actinide inventory, and (3) the ENDF/B-V to ENDF/B-VI cross section updates.

The LTR includes comparisons of the GENE and BNL NUREG/CR-6115 MCNP fluence predictions. Comparisons are provided for the $E > 1.0$ MeV fluence at the axial midplane for locations in the downcomer and on the vessel inner-wall.

2.3 Calculation of the BWR Neutron Dosimetry Benchmark Measurements

In order to provide a measurement benchmark for qualifying the DORT and MCNP calculational methodology, GENE has performed an in-reactor dosimetry benchmark experiment (References 4 and 5). The experiment included the irradiation of a set of passive dosimeters for one cycle in an operating (non-US) BWR. The measurements included Fe-54, Nb-93, and Ni-58 threshold dosimeters as well as U-238, Th-232 and Np-237 fission dosimeters. The dosimeters were located in the downcomer at three axial elevations, three azimuths and three radial locations. The dosimeter activation counting and related measurements were performed at the GE Vallecitos Nuclear Laboratory.

The neutron dosimetry provides a direct measurement of the activity (dps/gm) associated with the individual dosimeter-specific reactions. The measured dosimeter activities were converted to specific full power reaction rates (rps/nucleus) averaged over the period of irradiation. This conversion accounted for the physical characteristics of the sensor (e.g., weight of the target isotope in the sample), the operating history of the reactor, the energy response of the sensor (e.g., reaction cross section), decay of the target isotope, and in the case of fission dosimeters, the number of product atoms produced per reaction. In order to allow comparison of the measured and calculated dosimeter reaction rates, the measured reaction rates have also been corrected for target depletion.

The in-vessel dosimetry measurements were used to benchmark and validate the proposed calculational methodology. The validation included both DORT and MCNP calculations of the measured dosimeter reaction rates. The calculational models used in the prediction of the measurements are based on the proposed methodology described in Section 2.1. The models include a detailed representation of the peripheral fuel assemblies and the core internals, downcomer and vessel geometry. The DORT calculations employ a relatively fine (r, θ, z) spatial mesh and were carried out using an S_{12} quadrature and a P_3 expansion of the scattering cross sections.

The calculations of the dosimeter response measurements are used to determine the calculation-to-measurement data base used to validate the fluence calculation methods. The analysis of the C/M data indicates that: (1) the DORT calculations using an adjusted downcomer model result in a mean C/M value ranging from 0.9 to 1.1 for Fe-54 and Nb-93, and (2) the MCNP calculations result in C/M values ranging from 0.9 to 1.1.

3.0 EVALUATION

The review of the NEDC-32983P methodology focused on the details of the fluence calculation methods, their compliance with the guidance in RG 1.190 and the qualification of the methodology provided by the GENE C/M data-base. As a result of the review, several technical issues were identified which required additional information and clarification from GENE. Requests for additional information (RAI) were transmitted in References 9, 13 and 16. The GENE responses were provided in References 10-12, 14-15, and 17-18. This evaluation is based on the material included in the LTR and in the referenced GENE responses to the RAIs. The evaluation of the major issues raised during the review is summarized in the following.

3.1 Pressure Vessel Fluence Calculation Methodology

The DORT transport calculational model is constructed using plant-specific as-built dimensions and actual plant parameters whenever possible (response to RAI-5, Reference 10). The calculations use a fine spatial and angular mesh in both the (r, θ) and (r, z) calculations together with a detailed representation of the core internals, downcomer, and vessel geometry. The calculations employ an S_8 angular quadrature set and a P_3 scattering cross section expansion.

The proposed fluence methodology generally employs a best-estimate approach, however, certain conservative features have been retained from the traditional method. For example, in response to RAI-2 (Reference 10), GENE indicates that the core neutron source used in the

DORT transport calculation is based on the bundle-average power in the peripheral fuel bundles. This results in a conservatism in the fluence estimate because: (1) the fuel pins close to the core edge have reduced power because of neutron leakage from the core, and (2) the fuel pins close to the core edge provide the dominant contribution to the vessel fluence. The magnitude of the effect of using the bundle-average power rather than the pin-wise power distribution is calculated for the BWR pressure vessel fluence benchmark problem (Reference 3). In addition, in the response to RAI-2 (Reference 15), GENE indicated that this conservatism in the fluence calculation is applicable to all core designs. However, GENE has indicated in response to RAI-18 (Reference 10) that credit for this conservatism will be taken in determining the adjustment that must be applied to the calculated fluence to determine the best-estimate fluence value. Therefore, while the proposed current methodology includes this conservatism and over-predicts the fluence due to the use of bundle-average power, this conservatism is removed in the application of the methodology when the best-estimate fluence is determined, by a downward adjustment of the calculated fluence.

The nuclear cross section library used in the fluence transport calculations employs a P_3 Legendre expansion of the anisotropic cross sections. However, because of the relatively strong axial dependence of the void distribution in the core and the presence of the jet-pump and jet-pump riser arrangement in the downcomer, there was concern that the third order Legendre expansion may not be sufficiently detailed to accurately model the streaming and shadowing effects at the vessel inner-wall. In order to evaluate this effect, GENE has performed a series of sensitivity calculations using a P_5 expansion of the anisotropic scattering cross section. The results of these calculations are presented in the response to RAI-7 (Reference 10) and indicate that the effect of this approximation on the vessel fluence and dosimetry reaction rates is negligible.

3.2 Calculation of the BNL Pressure Vessel Fluence Benchmark Problem

The BNL pressure vessel fluence benchmark problem was calculated as part of the validation and testing of the NEDC-32983P fluence methodology. The calculations were carried out using the proposed GENE methodology (response to RAI-5, Reference 2) and were compared with the tabulated benchmark reference predictions. The analysis of the benchmark problem included a set of sensitivity calculations which evaluated and confirmed the validity of several modeling assumptions included in the methodology. The GENE and reference calculations of the vessel peak inner-wall fluence were found to be in good agreement.

In the proposed methodology, the DORT transport calculations are performed using a nuclear cross section set that has been collapsed by averaging the nuclear data over a multi-group energy structure. Following the guidance in RG 1.190 (Section-1.1.2.2), GENE tested and evaluated the averaging procedure used in collapsing the cross sections. The evaluation included a series of DORT transport calculations which were carried out for the BNL vessel fluence benchmark problem using several sets of collapsed cross sections. The results of these calculations are included in the GENE response to RAI-3 (Reference 10). Calculations were performed for a 26-group cross section set, a 44-group cross section set and a 47-group cross section set (calculated by BNL). Comparisons of the E > 1.0 MeV flux and the flux spectrum were made at the shroud, downcomer, surveillance capsule, vessel inner-wall, vessel quarter-thickness and vessel outer-wall locations. Based on these comparisons and additional calculations performed by GENE, it is concluded that the use of the collapsed cross section library introduces a bias into the fluence prediction (response to RAI-3, Reference 10). GENE has indicated in response to RAI-18 (Reference 10) that, in order to account for this approximation, the fluence calculated with the NEDC-32983P methodology will be adjusted to determine the best-estimate fluence value. Therefore, while the proposed current methodology includes this calculational bias due to the cross section averaging procedure, this bias will be removed when the best-estimate fluence is determined.

3.3 Calculation of the BWR Benchmark Dosimetry Measurements

The BWR neutron dosimetry experiment includes an extensive set of in-vessel fast and thermal neutron dosimeter measurements. The irradiation of the dosimeters was performed during a single cycle of operation at an operating BWR. The dosimeter activation and associated measurements were performed at the GE Vallecitos Nuclear Laboratory. The inferred reaction rates are proportional to the measured specific activities and include adjustments for the actual plant operating history and the decay of the reaction product isotope. The reaction rates were used to construct the C/M benchmark data base and determine the calculational bias and uncertainty.

The initial analysis of the BWR neutron dosimetry experiment did not include C/M comparisons for the dosimetry measurements at the 71° azimuth. However, in response to RAI-9 (Reference 15), GENE has updated the C/M data base to include this data. This additional C/M data is generally consistent with data taken at 4° and 20°. In order to allow valid benchmarking C/M comparisons of the calculations and the dosimetry experiment measurements, reliable estimates of the uncertainty in the dosimetry measurements are required. In response to RAI-11 (Reference 11), GENE has provided the uncertainty analysis for the dosimetry experiment measurements. The statistical uncertainty in the specific dosimeter activity measurement is provided for both the fast and thermal dosimeters. The measurement uncertainty resulting from the uncertainty in the capsule location is based on: (1) the mechanical tolerance for capsule displacement, and (2) the sensitivity of the dosimeter response to capsule displacement. Since the spatial variation of the fast and thermal flux (and associated displacement sensitivity) is different, the measurement uncertainty due to capsule displacement is determined for both the fast and thermal dosimeters.

In addition to the BWR neutron dosimetry experiment, the GENE dosimetry benchmark data base includes a set of surveillance capsule flux measurements. This surveillance capsule data base includes a range of plant measurements that have been made over the past decade. The activity measurements were carried out using a set of standard fast neutron threshold dosimeters. The GE Vallecitos Nuclear Laboratory analyzed the activity measurements and determined the analysis uncertainty. The activation measurement is converted to flux using a dosimeter specific cross section determined by a series of specially controlled experiments. In response to RAI-3 (Reference 14), GENE has indicated that the methods used to analyze these surveillance dosimetry measurements are compliant with the ASTM standards for measuring fast-neutron reaction rates by radioactivation of iron, copper and nickel; ASTM E-263-93 (Reference 19), ASTM E-523-92 (Reference 20) and ASTM E-264-92 (Reference 21), respectively.

3.4 C/M Comparisons and Uncertainty Analysis

The qualification of the NEDC-32983P pressure vessel neutron fluence methodology includes comparisons of fluence calculations and measurements for: (1) the operating reactor benchmark dosimetry experiment and (2) the BWR surveillance capsule dosimetry measurements. The methods benchmarking is based on both the BWR dosimetry experiment and the surveillance capsule measurements. The benchmark experiment measurements include a set of fast neutron threshold dosimeters located in the downcomer at three axial elevations, three azimuths and three radial locations. The BWR surveillance measurements are for capsules located at various locations in the downcomer including within the shadow and the penumbra of the jet pumps and jet pump risers. The dosimetry experiment provides a continuous fluence measurement during the single cycle of irradiation, while the surveillance capsule measurements provide a continuous fluence measurement from initial startup to the time of capsule removal which represent a variety of irradiation time intervals. These operating reactor measurements provide an indication of the effect of the as-built geometry and material compositions on the fluence calculations. The benchmarking is based on the calculation-to-measurement (C/M) comparisons of the measured reaction rates. The measurements provide a number of C/M comparisons and a statistical estimate of the calculational bias and uncertainty.

The benchmark experiment comparisons are made for each location as a function of dosimeter type (e.g., Fe-54 and Nb-93). In the response to RAIs 10 and 14 (Reference 12), GENE has provided the C/M ratios and analysis for the dosimetry benchmark experiment. In addition to the Fe-54 and Nb-93 bare capsule dosimeters included in the LTR, Ni-58 and Nb-93 shielded capsules were also evaluated. The C/M analysis for the dosimetry benchmark experiment indicates that the calculations are within 20 percent (one-σ) for the vessel measurements.

In the responses to RAI-17 (Reference 12) and RAI-7 (Reference 15), GENE has provided a statistical analysis of the C/M comparisons for the BWR capsule surveillance measurements. The analysis included in the responses to RAIs 17 and 18 (Reference 12) and RAI-7 (Reference 15) indicates that the proposed methodology is biased relative to the measurements. The C/M bias and its uncertainty have been determined using statistical techniques. In the proposed methodology, the best-estimate fluence is determined by applying the C/M bias to the calculated fluence. In addition, GENE has indicated in response to RAI-8 (Reference 15) that as new measurements become available these comparisons will be updated. If necessary, the bias and its uncertainty will be updated and the adjustment to the calculated fluence will be revised.

In order to provide an independent estimate of the bias and uncertainty in the NEDC-32983P fluence calculational methodology, GENE has performed an analytic uncertainty estimate. The significant sources of bias/uncertainty were identified by a set of DORT fluence sensitivity calculations. These calculations concerned the treatment of the nuclear cross section data, core neutron source, angular quadrature, and geometrical representation of the downcomer. In addition, in response to RAI-6 (Reference 15), GENE has included the effect of the BWR fuel bundle nodal and pin-wise power distribution uncertainty on the calculated fluence. Estimates of the important uncertainty contributors were made and the effect of these uncertainties was propagated through the fluence calculation using the calculated sensitivities. In the response to

RAI-18 (Reference 12) and RAI-6 (Reference 15), the analytically determined fluence calculational uncertainty is shown to be less than 20 percent.

The significant sources of calculational bias were determined to be: (1) the effect of using the bundle-average power rather than the pin-wise power distribution in the peripheral fuel bundles, and (2) the effect of using a specific flux-averaged multi-group cross section set. In the response to RAI-18 (Reference 12), the overall fluence calculational bias is determined analytically as a combination of these individual components. The bias determined using the analytic method was found to be slightly less but well within the uncertainty range of the bias determined based on the surveillance dosimetry measurements. In the conclusion of the response to RAI-7 (Reference 15), GENE stated that the calculational bias based on the dosimetry measurements will be applied to the fluence calculated using the NEDC-32983P fluence methodology.

While the uncertainty analysis based on the surveillance dosimetry C/M comparisons is generally consistent with the analytic uncertainty, it is noted that several substantial adjustments are required to account for approximations made in the calculations of the surveillance data. In addition the uncertainty in the fluence adjustment is not substantially smaller than the adjustment itself. Therefore, in order to provide additional confidence in the benchmarking of the proposed fluence methodology, within three years GENE is required to perform predictive calculations of at least four additional BWR capsule dosimetry activity measurements. These calculations should be submitted to the NRC staff prior to the completion of the measurements. After the measurements are completed, comparisons of the measurements and calculations should also be submitted to the NRC. If the C/M comparisons are not consistent with the proposed NEDC-32983P fluence methodology and supporting benchmark uncertainty analysis, the necessary revisions to the uncertainty analysis and methodology should be provided in the submittal. This requirement was discussed and agreed upon with GENE in a NRC/GENE/BNL conference call on June 25, 2001.

3.5 Core Shroud

In addition to the calculation of pressure vessel fluence, GENE has indicated that the proposed fluence methodology may be required for material evaluations of the core shroud. GENE has described the shroud fluence calculational procedure and provided an analytic estimate of the calculational uncertainty in response to RAI-8 (Reference 17).

As benchmarking for the shroud fluence calculation, in Figure 5-4 of the LTR and in the response to RAI-8 (Reference 17), GENE has provided comparisons of reaction rates calculated with the proposed methodology and reaction rates determined from measurements for capsules located close to the shroud. No direct shroud data were provided. The benchmark experiment C/M comparisons for the shroud indicate a conservative bias and a systematic over-prediction of the measurement data. However, review of this data indicates that the C/M comparisons for these dosimeters include large differences that are outside the expected calculation and measurement uncertainties. Consequently, because the bias is based on a single experiment and there is no surveillance data to confirm this result, this conservatism is not considered sufficiently reliable to reduce the calculated shroud fluence.

However, shroud fluence values are used mainly for the estimation of shroud crack growth propagation rates. The phenomenon is associated with a threshold fluence value. Therefore, the staff finds the proposed method acceptable for shroud fluence calculations provided that: (1) the estimates are limited within the beltline region, and (2) the bias is not deducted from the calculated value. To provide additional confidence to the predicted shroud fluence, GENE is required within three years from the approval of this methodology to perform and provide to the staff additional dosimetry analysis, directly related to the shroud, demonstrating the capability of this method.

4.0 SUMMARY AND LIMITATIONS

The staff reviewed NEDC-32983P entitled, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," and supporting documentation provided in References 10-12, 14-15 and 17-18. Based on this review, it is concluded that the NEDC-32983P methodology provides an acceptable best-estimate prediction of the pressure vessel neutron fluence for US BWR plants. As discussed in Section 3.4 of this SE, the best-estimate vessel fluence prediction is determined by the application of the calculated-bias adjustment to the fluence estimate using the NEDC-32983P fluence methodology.

However, this acceptance is subject to the following limitations and requirements (Sections 3.4 and 3.5):

(1) Within three years from the day of the approval of this methodology, GENE will perform predictive calculations of at least four additional BWR surveillance capsule dosimetry measurements which will be submitted to the staff before initiation of the measurements.

(2) Comparisons of the measurements and calculations will also be submitted to the NRC.

(3) Shroud fluence estimates will be limited to the beltline region, without bias adjustment.

(4) GENE will perform dosimetry analysis to confirm and remove the conservatism in the shroud fluence calculations.

(5) Revisions to the fluence methodology and supporting uncertainty analysis will be provided, if the C/M comparisons (for the additional analysis for the vessel and the shroud) are not consistent with the NEDC-32983P fluence methodology.

5.0 REFERENCES

1. "Submittal of GE Proprietary Document NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," Letter, J. F. Klapproth (GENE) to USNRC, dated September 1, 2000.

2. Office of Nuclear Regulatory Research, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Regulatory Guide 1.190, U.S. Nuclear Regulatory Commission, April 2001.

3. "Pressure Vessel Fluence Calculation Benchmark Problems and Solutions," NUREG/CR-6115 (BNL NUREG-52395), Brookhaven National Laboratory (to be published).

4. Terhune, J. H., Sitaraman, S., Higgins, J., P., Chiang, R-T., and Asano, K., "Neutron and Gamma Spectra in the BWR- Phase 1 Experimental and Computational Methods," Proceedings of the Fifth International Conference on Nuclear Engineering, Nice, France, ICONE5-2020, May 26-30, 1997.

5. Sitaraman, S., Chiang, R-T. , Asano, K., and Koyabu, K., "Benchmark for a 3D Monte Carlo Boiling Water Reactor Fluence Computational Package - MF3D," International Conference on Advanced Monte Carlo for Radiation Physics, Particle Transport Simulation and Applications, Lisbon, Portugal, October 23-26, 2000.

6. "DORT, Two-Dimensional Discrete Ordinates Transport Code," RSIC Computer Code Collection, CCC-484, Oak Ridge National Laboratory, 1988.

7. J. F. Breismeister (Ed.), "MCNP – A General Monte Carlo N-Particle Transport Code, Version 4A," LA-12625, Los Alamos National Laboratory, March 1994.

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Date: September 14, 2001