October 21, 2005

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: DRAFT SAFETY EVALUATION FOR WESTINGHOUSE OWNERS GROUP TOPICAL REPORT WCAP-16083-NP, REVISION 0, "BENCHMARK TESTING OF THE FERRET CODE FOR LEAST SQUARES EVALUATION OF LIGHT WATER REACTOR DOSIMETRY" (TAC NO. MC3974)

Dear Mr. Bischoff:

By letter dated July 30, 2004, as supplemented by letter dated March 30, 2005, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-16083-NP, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

The NRC staff has completed the review of WCAP-16083-NP, Revision 0, and has determined that the proposed methodology in WCAP-16083 satisfies the guidance in Regulatory Guide 1.190, adopts the recommendations regarding dosimetry practices from several American Society for Testing and Materials standards, and it has been benchmarked against the National Institute of Standards and Technology fission sources and against the acceptable dosimetry measurements. Therefore, this methodology is acceptable for use in licensing actions regarding light-water reactor dosimetry, subject to the limitation described in the attached draft safety evaluation (SE).

Twenty working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The NRC staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the NRC staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact Girija Shukla at (301) 415-8439.

Sincerely,

/RA/
Daniel S. Collins, Acting Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Draft Safety Evaluation

cc w/encl:
Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16083-NP, REVISION 0, "BENCHMARK TESTING OF THE FERRET CODE FOR LEAST SQUARES EVALUATION OF LIGHT WATER REACTOR DOSIMETRY" WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated July 30, 2004, as supplemented by letter dated March 30, 2005 (References 1 and 2, Agencywide Documents Access and Management System Accession nos. ML042160524, and ML050910119, respectively), the Westinghouse Owners Group (WOG) submitted Topical Report WCAP-16083-NP, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," to the Nuclear Regulatory Commission (NRC) staff for review.

The methodology proposed in WCAP-16083 consists of three phases: (1) collection of a data base of benchmarked plant-specific neutron transport calculations and corresponding dosimetry measurements at in-vessel and ex-vessel locations, (2) a least squares analysis involving the calculated and measured data, and (3) use of the results to demonstrate consistency of measured and calculated values and to validate calculated values at locations on the vessel inside diameter. The least squares adjustment method uses neutron spectra adjustment, dosimeter spectral coverage, transport calculation uncertainties, measured reaction rates, and dosimeter cross sections and their uncertainties. This approach is endorsed by and is summarized in American Society of Testing and Materials (ASTM) Standard E 944-02 (Reference 3).

The purpose of this review is to describe the code, establish whether the method adheres to the guidance in Regulatory Guide (RG) 1.190 (Reference 4), examine the validation of the code and evaluate the acceptability of the proposed method in light water reactor (LWR) licensing actions.

2.0 REGULATORY EVALUATION

The basis for this review is RG 1.190 (Reference 4) that is based on General Design Criteria 14, 30, and 31, and describes the attributes of neutron transport methodologies which are acceptable to the NRC staff. RG 1.190 specifies that the neutron transport methods should be benchmarked to a statistically significant data base of measurement-to-calculation ratios (M/C) and that existing bias and uncertainties be estimated. In addition, the RG allows the use of suitably weighted averages of the M/C values.

3.0 <u>SUMMARY OF THE FERRET LEAST-SQUARES ADJUSTMENT METHODOLOGY</u>

3.1 Background

The proposed least squares adjustment (LSA) method combines measurement data with corresponding neutron transport calculations to establish a best estimate spectrum and an estimate of the applicable uncertainties at the location of the measurement. The spectrum is then used to calculate best estimate values of exposure quantities, such as activation rates, fluence, and iron displacements per atom. The FERRET code, which is a least squares adjustment, has been applied successfully in many reactor vessel applications. The ASTM promulgated the standard E 944-02 to address the application of neutron spectrum adjustment methods to reactor surveillance dosimetry. It is assumed that neutron transport is using the discrete elements method as in the DORT Code (Reference 5).

3.2 Application of the Methodology

The general objective of an LSA method is to reconcile measured and calculated reaction rates, dosimetry and transport cross sections, and calculated neutron energy spectra within their corresponding uncertainties. In general, the following expression relates reaction rate R_i to neutron energy spectrum φ_g, and to dosimeter (group) reaction cross section σ_{ig}, each with a corresponding uncertainty δ:

$$R_i \pm \delta_{Ri} = 3\ (\sigma_{ig} \pm \delta_{\sigma ig})\ (\varphi_g + \delta_{\varphi g})$$

Application of the LSA method requires the following information for a specific measurement: (1) a calculated spectrum and its uncertainty, (2) dosimeter measured reaction rate and uncertainty, and (3) dosimetry reaction cross sections and their uncertainty. The plant-specific neutron transport calculations yielding the neutron energy spectrum should follow the guidance in RG 1.190.

3.3 Neutron Transport Calculations and Uncertainty

The neutron transport calculation forms the basis for a reliable LSA. The flux synthesis method is used to calculate the three-dimensional neutron flux distribution $\varphi(r, \theta, z)$ as follows:

$$\varphi(r, \theta, z) = \{(\varphi(r, \theta) * \varphi(r, z)\}/\varphi(r)$$

where $\varphi(r, \theta)$, $\varphi(r, z)$ and $\varphi(r)$ are the azimuthal, axial and radial flux distributions, respectively.

The WOG is using the DORT (Reference 5) discrete ordinates code and the BUGLE-96 (Reference 6) cross section library. An anisotropic scattering is treated with a minimum of a P_3 approximation and an S_8 minimum angular quadrature. As stated previously, transport calculations follow the guidance in RG 1.190. P_3 and S_8 are discussed in some detail in RG 1.190.

3.4 Geometric Modeling

In developing the geometrical representation of the vessel, core, and internal components the effort is to use "as-built" dimensions where available. Water temperatures (and thus water densities) are assumed at full power. The core is represented as a mixture of fuel, cladding, water, and structural materials at temperatures representing full-power operation. The choice of mesh size in the axial, radial, and azimuthal directions are chosen to achieve convergence in the inner iterations. In general, smaller intervals are chosen in areas where large flux gradients are anticipated. Normally, quarter core or octant core symmetry is applied. The core baffle, the former plates, and the thermal shield are represented as individual components.

3.5 Neutron Source

The source distribution is obtained from pin-wise power distribution from the two outer row fuel assemblies. The fuel isotopic composition is accounted for as a weighting factor in the power-to-neutron conversion. The (r, θ) geometry transposition to (x, y) uses an area weighting to assign source strength to each (x, y) cell from the corresponding (r, θ) cell(s).

3.6 Validation of the Transport Calculation

The WOG used the transport method described in WCAP-14040-A (Reference 7) that has been approved by the NRC staff. The validation was based on the guidance in RG 1.190 and included comparison to the Oak Ridge Pool Critical Assembly (PCA), the H. B. Robinson dosimetry benchmark experiment, an experimental data base consisting of a large number of surveillance capsules from a variety of operating plants, and an analytical sensitivity study addressing the major uncertainty components.

The WCAP-16083 validation includes three stages: (1) methods' validation addressing the adequacy of the transport calculation and associated dosimetry and cross sections, (2) validation of uncertainties that are methods-related, and (3) validation addressing uncertainties that are related to lack of knowledge of code input parameters. The overall calculational uncertainty is established from the above components.

3.7 Uncertainty Input to LSA

The neutron energy spectrum in each measurement location is input as an absolute value. Spectrum uncertainty is obtained from plant-specific transport calculations also at the location of the measurement. The spectrum input uncertainties should be consistent with the benchmarking results discussed in Section 3.6. The uncertainty matrix is constructed from the following relationship:

$$M_{g'g} = R_n^2 + R_g * R_{g'} * P_{g'g}$$

where R_n is the overall fractional normalization uncertainty, R_g and $R_{g'}$ are groupwise uncertainties, and $P_{g'g}$ is a group correlation matrix. Analytic expressions for $P_{g'g}$ are also provided. The normalization uncertainty is related to the magnitude of the spectrum, while the groupwise uncertainties are related to the shape of the spectrum. WCAP-16083 provides specific numerical values for the uncertainties.

3.8 Reaction Rate Measurement and Uncertainties

WCAP-16083 lists the standard dosimeters used by The WOG: Cu-63(n,α)Co-60, Ti-46(n,p)Sc-46, Fe-54(n,p)Mn-54, Ni-58(n,p)Co-58, U-238(n,f) fp (Cd covered), Np-237(n,f) fp (Cd covered), Co-59(n,γ)Co-60 (with and without Cd cover). This dosimeter set provides adequate spectral coverage. WCAP-16083 lists the ASTM standards relevant to the recommended practice for the use of these monitors. The analytical expression to calculate the average dosimeter activation for a given power level from the measured activation rate is given. The section concludes with values of specific uncertainties and their justification.

3.9 Dosimetry Cross Sections and Uncertainties

The activation cross sections and the associated uncertainties are obtained from the SNLRML library (Reference 8) that is based on the ENDF/B-VI file.

4.0 TESTING OF THE FERRET PROCESSING PROCEDURES

As noted above, FERRET combines the dosimeter reaction rate measurements with the results of the neutron transport calculations, dosimetry reaction cross sections, and neutron spectra to calculate a best estimate fast neutron flux (E > 1.0 MeV) at the location of the measurement. The process is divided into two steps: (1) processing of the calculated spectra and dosimetry cross sections and (2) application of the FERRET algorithm. Each of the steps is individually tested as outlined in the following paragraphs.

4.1 Data Comparison in the National Institute of Standards and Technology (NIST) U-235 Fission Field

The SNLRML cross sections are collapsed 53 energy groups using the calculated energy spectrum as a weighting function. The FERRET report used the data in ASTM report E261-98 (Reference 9) fission spectrum averaged cross sections applicable to U-235 and Cf-252 spectra. The section lists numerous other comparisons with existing data to conclude that the SNLRML library and the FERRET processing result in accurate cross section values.

4.2 Evaluation of the PCA Simulator Benchmark

RG 1.190 recommends benchmarking to the results of the PCA (Reference 10). In the past, PCA has been analyzed by several researchers using least squares codes. The WOG updated the existing calculations using updated cross sections. Comparison of the measured values to the updated calculated results demonstrates good-to-excellent agreement after the adjustment. In addition, comparisons indicate consistency of the FERRET results from other analyses' methods and for all the measured locations.

4.3 Evaluation of the H.B. Robinson Benchmark

The H.B. Robinson (Reference 11) vessel dosimetry measurements were also used in the FERRET benchmark. The transport calculations were carried out using the BUGLE-96 library based on the ENDF/B-VI file, the P_3 anisotropic scattering, and the S_8 angular quadrature approximations. The Robinson measurements consist of in-vessel and ex-vessel dosimetry. The FERRET adjustment for both sets is very small and consistent with the uncertainty bounds.

5.0 FERRET SENSITIVITY STUDIES

The purpose of the sensitivity study is to evaluate the impact of the spectral uncertainty and of the foil composition on the LSA.

5.1 Composition of the Multiple Foil Sensor Set

In this case, the spectral uncertainties were held constant as well as the uncertainties associated with the reaction rates. The base case consisted of a set of six dosimeters (Cu, Ti, Fe, Ni, U-238, and Np-237). Ten additional cases were constructed by dropping one or more dosimeters from the base case and calculating the adjusted/calculated (A/C) ratio. These were then compared to the base case. The results indicate that for minimum uncertainty the dosimeter set should include Fe, U-238, and Np-237 foils.

5.2 Input Uncertainties

In this part of the study the reaction rate and the spectrum uncertainties were assigned high, medium, and low values. Considering the medium-medium case as the base-case the magnitude of the adjusted flux changes very little. However, the associated uncertainty changed considerably more, as expected.

6.0 TECHNICAL EVALUATION

6.1 Introduction and Historical Note

Least squares adjustments have been applied for many years in dosimetry analyses. The ASTM Standard E 944 (Reference 3) includes an extensive list of codes and methods that have been adopted for dosimetry problems. FERRET, in particular, which was developed at the Hanford Engineering Development Laboratory (HEDL), has been used in the liquid metal fast breeder reactor and the NRC-sponsored LWR pressure vessel surveillance dosimetry improvement program (LWR-PV-SDIP). The PCA benchmark experiment was part of the LWR-PV-SDIP program.

In the past, issues have been raised regarding the consistency of the M/C data bases for LWR applications. The WOG stated that variations due to neutron energies, dosimeter locations, transport and activation cross sections, and time periods have been removed.

As stated earlier, application of the FERRET code requires three types of input information: (1) calculated neutron energy spectrum and uncertainty, (2) measured reaction rates and uncertainties, and (3) energy-dependent dosimetry reaction cross sections. The following sections evaluate each input type.

6.2 Neutron Transport Calculations

Although the required information is the neutron spectra at the location of the measurements, an accurate neutron transport calculation is needed to obtain the spectra at given locations.

The method is based on the synthesis technique that combines two two-dimensional solutions in (r, θ) and (r, z) to produce a three-dimensional flux:

$$\varphi(r, \theta, z) = [(\varphi(r, \theta) * (\varphi(r, z)]/[\varphi(r)]$$

The transport calculation is carried out using the discrete ordinates, finite difference code DORT, using the BUGLE-96 cross sections, derived from the ENDF/B-VI file. This calculation adheres to the guidance in RG 1.190 and, therefore, it is acceptable.

6.3 Geometric Modeling

The geometric modeling should be designed to preserve the physical accuracy of the material regions. This is accomplished by using the appropriate number of mesh points. The description of this model states that up to 250 radial points, 110 azimuthal, and 150 axial points may be used. The point distribution is judicious by accommodating areas of expected high flux gradients and high total cross section. Also, the inner iteration convergence criterion is set at 0.001. All of these features agree with the guidance in RG 1.190, therefore, the proposed geometrical model is acceptable.

6.4 Core Source

Because neutron sources are volumetric and in (x, y) geometry, their transposition to (r, θ) geometry must preserve the fuel volume. In addition, to assure that the energy spectrum is correct the isotopic composition of the fissionable nuclei must be represented correctly for the irradiation period represented in the calculation. Finally, the number of neutrons released per fission is also a function of the isotopic composition of the fissionable nuclei. The proposed method is designed to maintain the source volume and estimate the fissionable nuclei through burnup. The review indicates that the source calculation is acceptable because its transposition maintains the volume and accounts for its isotopic composition assuring correctness of the energy spectrum and the number of neutrons produced per fission.

6.5 Validation of the Transport Calculation

The validation process is based on the guidance in RG 1.190 and includes comparisons with the PCA benchmark experiment, the H. B. Robinson measurements, an analytic sensitivity study, and comparison to an extensive data base consisting of surveillance capsule measurements from operating plants. The validation addresses the adequacy of the transport calculational method, method related uncertainties, and uncertainties due to imperfect knowledge of the input data.

The results of the validation are well within the 20 percent (1σ) uncertainty prescribed in RG 1.190. In addition, the transport methodology is based on WCAP-14040-A that has been approved by the NRC. Because the methodology has been approved, the validation process is as prescribed by RG 1.190 and, the results are within recommended limits, the NRC staff finds the validation acceptable.

6.6 Uncertainty Input to the Least-squares Adjustment

The adjustment algorithm is based on the absolute value of the neutron spectrum at the location of the measurement. The input is the spectrum uncertainty and is expressed as an uncertainty matrix that contains the normalization uncertainty related to the magnitude of the spectrum and groupwise uncertainties. The values of the normalization and groupwise uncertainties presented in WCAP-16083 are within the range of similar values in the literature and well within the uncertainties specified in the transport solution, therefore, the proposed method is acceptable.

6.7 Reaction Rate Measurement and Uncertainties

Flux measurements in operating plants are accomplished with a set of dosimeters that assures good spectral coverage. Such a set was identified in Section 3.8 above. ASTM standards (E series) outline methods to optimize the efficiency and to maximize the accuracy of the dosimeter measurements. WCAP-16083-NP states that the applicable standard is used for each dosimeter. In addition to the threshold detectors (as listed in Section 3.8), solid state track recorders that directly measure total (fluence) exposure are also mentioned in WCAP-16803-NP. Conventional dosimeters measure activation that is converted analytically to an irradiation rate and subsequently to fluence. WCAP-16083-NP outlines the special procedures required for the fission dosimeters in particular. WCAP-16803-NP outlines several tests that demonstrate the historical improvement and evolution of dosimetry measurement accuracy. The values of the (1σ) uncertainties for the dosimeter set in Section 3.8 are similar to those found in the literature. In summary, the NRC staff finds the reaction rate measurement uncertainty to be acceptable because the measurement process followed accepted standard procedures, because they have been benchmarked to existing standards, and because the values are comparable to those found in the literature.

6.8 Dosimetry Cross Sections and Uncertainty

Section 6.6 dealt with dosimeter uncertainties originating in the counting process. This section presents dosimeter activation cross section uncertainties. The uncertainties for the dosimeter set presented in Section 3.8 are part of the SNLRML library (Reference 8). These have been compiled from the most recent data and extensively tested for consistency and accuracy. Because the SNLRML cross sections and their uncertainties are in general use for dosimetry work and because they have been subjected to extensive testing, they are acceptable for the proposed least squares adjustment for FERRET.

6.9 Data Comparison in the NIST U-235 Fission Field

Measurements of the dosimeter cross sections and their uncertainties are recorded in ASTM E 261-98, "Standard Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Technique" (Reference 11). Comparisons of calculated and measured values of the cross sections in the U-235 spectrum and the same from the PCA measurements are shown in tabular form within ASTM E 261-98. Uncertainties documented in ASTM E261-98 are within the (1σ) range. The calculational method employed in ASTM E 261-98 is the same as that used by The WOG, therefore, the results are applicable. The same data are also available for the Cf-252 spectrum with similar results. These results support the claim for the value of the uncertainties and their suitability for the least squares analysis in FERRET and, therefore, the results are acceptable.

6.10 Evaluation of the PCA Simulator Benchmark

RG 1.190 recommends the use of the results from the PCA experiment to compare and benchmark transport calculations and associated uncertainties. WCAP-16083-NP presents transport calculations for positions A_1 to A_7 representing the inside surface of the thermal shield to the outside of the pressure vessel, including the point inside the vessel thickness. The measured to calculated ratios fall in the range of 0.91 to 1.05. The adjusted values in terms of measured to adjusted ratios (M/A) are in the range of 0.94 to 1.06. The differences, the adjustments, and the uncertainties are small and consistent with the uncertainty bounds for the reaction rates and the neutron flux. The same conclusion is reached by analyzing similar calculations on PCA performed by HEDL, Oak Ridge National Laboratory (ORNL), and others. In summary, analyses of the PCA benchmark experiment using the FERRET code yielded results that are consistent with prescribed uncertainty bounds. The uncertainty bounds become smaller when adjusted using the FERRET code. This supports the use of the FERRET code.

6.11 Evaluation of the H. B. Robinson Benchmark

This is a case of laboratory quality surveillance applied to an operating plant. The analysis and evaluation were sponsored by the NRC, were performed by ORNL, and are documented in NUREG/CR-6453 (Reference 9). A discrete ordinates code was used with the BUGLE-96 cross sections that are based on the ENDF/B-VI file. The calculations used the P_3 inelastic scattering and the S_8 angular quadrature approximations. Review of the M/C ratios (before adjustment) indicates that they fall in the range of 0.95 to 1.11. The M/A ratios adjusted individual dosimeter values fall in the range of 0.96 to 1.09. The FERRET code adjustment procedure reduced the uncertainty.

6.12 FERRET Sensitivity Studies

Two studies examine the relative position of the threshold dosimeters to the in-vessel and ex-vessel spectrum and the effect of the composition of the foil set in the accuracy of the results, assuming that the full set of detectors results in the most accurate results. These studies are not a necessary part of the adjustment procedure but are instructive to the dosimetry analyst.

The first exercise indicates that in order to validate a calculation of the neutron flux, spectral weighting should be included in the calculations. The other indicates that to minimize the uncertainty using dosimeter measurements the dosimeter set should as a minimum include Fe, U-235, and NP-237.

6.13 Conditions for the Applicability of Least-squares Adjustment

From the above discussion it is apparent that to successfully employ LSA, the measured and calculated values must be within their own uncertainty bounds. Should this not be the case, both measured and calculated values must be re-examined for possible errors and, if they cannot be found, the particular values should be disqualified. WCAP-16803-NP states that: (1) in the past, data base consistency issues have been raised and (2) that the data base used in the FERRET benchmarking meets this condition.

7.0 CONCLUSIONS AND LIMITATION

The WOG submitted the FERRET code for NRC staff review and approval. FERRET is a least squares adjustment code using calculated spectra weighting to minimize calculated value uncertainties. In addition to the spectra, it also uses measured reaction rates and dosimetry cross sections and associated uncertainties. The adjusted neutron fluxes could be used to form a data base to validate neutron transport calculations in accordance with the guidance in RG 1.190. The results of the FERRET adjustment have been benchmarked by comparison to measurements in NIST-calibrated fission sources, the PCA simulated benchmark experiment, and the H. B. Robinson vessel dosimetry benchmark experiment. The transport calculation and the dosimetry cross sections adhere to the guidance in RG 1.190.

For the reasons stated above, the NRC staff finds that the FERRET code is acceptable to be referenced in operating plant licensing actions subject to the following limitation:

- LSA is acceptable if the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry-measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values should be disqualified.

8.0 REFERENCES

1. Letter from F.P. Schiffley II, Westinghouse Owners Group, to U.S. Nuclear Regulatory Commission, "Transmission of WCAP-16083-NP, Revision 0, 'Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,' " July 30, 2004.

2. Letter from F.P. Schiffley II, Westinghouse Owners Group, to U.S. Nuclear Regulatory Commission, "Revision to WCAP-16083NP, Revision 0, 'Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,' " March 30, 2005, and Letter from S. Anderson Westinghouse Owners Group to Lambros Lois, U.S. Nuclear Regulatory Commission “Historical Perspective on Reactor Dosimetry Data Bases,” August 22, 2005.

3. ASTM Standard E 944-02, "Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance," Annual Book of ASTM Standards, Section 12, Volume 12.02, 2003.

4. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, March 2001.

5. DOORS 3.1, "One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Safety Information Computation Center, Computer Code Collection CCC-650, Oak Ridge National Laboratory (ORNL) August 1996.

6. BUGLE-96, "Coupled 47 Neutron, 20 Gamma Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Shielding Information Center Data Library Collection DLC-185, Oak Ridge National Laboratory, March 1996.

7. WCAP-14040-A, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," by J. Andrachek, et. al., Westinghouse Owners Group, April 2002.

8. Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross Section Compendium," Radiation Shielding Information Center, Oak Ridge National Laboratory, July 1994.

9. ASTM E261-98, "Standard Practice for Determining Neutron Fluence, Fluence rate, and Spectra by Radioactivation Technique," ASTM Standards, Section 12, Volume 12.02, 2003.

10. NUREG/CR-6454, "Pool Critical Assembly Pressure Vessel Facility Benchmark," by I. Remek and F. B. K. Kam, Oak Ridge National Laboratory and U.S. Nuclear Regulatory Commission, July 1997.

11. NUREG/CR-6453, "H. B. Robinson Pressure Vessel Benchmark," by I. Remek and F. B. K. Kam, Oak Ridge National Laboratory and U.S. Regulatory Commission, February 1998.

Principal Contributor: Lambros Lois

Date: October 21, 2005