CTS-L/R-97-1

INDEPENDENT REVIEW

IS THE CONSUMERS ENERGY METHOD (WESTINGHOUSE METHOD) OF DETERMINING THE PALISADES NUCLEAR PLANT'S BEST ESTIMATE FLUENCE BY COMBINING TRANSPORT CALCULATION AND DOSIMETRY MEASUREMENTS TECHNICALLY SOUND AND DOES IT MEET THE INTENT OF THE PTS RULE ?

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Prepared by: W. N. McElroy

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by

Consultants & Technology Services

C **T** *S*

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- 3) DRAFT SECTION 5.0 FOR ANS-19.10 STANDARD; ATTACHMENT 3 of [Lo97].

- 4) ASTM SUBCOMMITTEE E10.05 TASK GROUP E10.05.04 REPORT ON E706 MASTER MATRIX LWR SURVEILLANCE STANDARDS; Dated March 30, 1996.

- 5) LWR PV SDIP: PCA EXPERIMENTS AND BLIND TEST; NUREG/CR-1861, July 1981. (SELECTED PAGES)

- 6) LWR PV SDIP PROGRAM: PCA EXPERIMENTS, BLIND TEST AND PHYSICS-DOSIMETRY SUPPORT FOR THE PSF EXPERIMENTS; NUREG/CR-3318, SEPTEMBER 1984. (SELECTED PAGES)

- 7) LWR PV SDIP PROGRAM: PSF PHYSICS-DOSIMETRY PROGRAM; NUREG/CR-3320, VOL. 3, OCTOBER 1987. (SELECTED PAGES)

- 8) LWR PV SDIP PROGRAM: LWR POWER REACTOR SURVEILLANCE PHYSICS-DOSIMETRY DATA BASE COMPENDIUM, 1987 UPDATE; NUREG/CR-3319, AUGUST 1985 & APRIL 1987. (SELECTED PAGES)

- 9) PROCEEDINGS OF FIRST ASTM-EURATOM SYMPOSIUM ON REACTOR DOSIMETRY, EUR 5667 e/f, PART 1, 1977. FIRST PAPER: NEUTRON ENVIRONMENTAL CHARACTERIZATION REQUIREMENTS FOR REACTOR FUELS AND MATERIALS DEVELOPMENT AND SURVEILLANCE PROGRAMS.

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- 12) J.R. WORSHAM'S PAPER ON "BIASED FLUENCES IN THE CHARPY EMBRITTLEMENT DATABASE," DISTRIBUTED TO ATTENDEES OF THE 9TH ASTM-EUROPEAN SYMPOSIUM ON REACTOR DOSIMETRY, HELD IN PRAGUE, CZECH REPUBLIC, SEPTEMBER 2-6, 1996

ABOUT THE AUTHOR

William N. Mc Elroy

Dr. McElroy has more then 45 years background experience in the nuclear physics and engineering, reactor development, standards and operations fields. He is a Fellow of ASTM and present or past member of ANS, American Association for the Advancement of Science, National Academy of Science/ National Academy of Engineering/National Research Council Evaluation Panel for Center for Radiation Research at NBS, Brookhaven Cross Section Evaluation Working Group (CSEWG), IAEA Working Group on Reactor Radiation Measurements, Phi Lambda Upsilon, Sigma Xi, Sigma Pi Sigma, and Phi Beta Kappa, and has extensive publications and holds several patents related to his work experience.

He is the Executive Chairman of ASTM Subcommittee E10.05 on Nuclear Radiation Metrology, Past Vice Chairman of ASTM Subcommittee E10.10 on Matrix to Standards for Nuclear Systems Technology, and the Past ASTM Executive Chairman of the ASTM-EURATOM Symposia on Reactor Dosimetry. He has been an ASTM representative on the NUMARC\NUPLEX Subcommittee on Codes and Standards and the ASME Board of Nuclear Codes and Standards Steering Committee on PLEX. He is responsible for the coordination, direction, and preparation of the ASTM E706 Master Matrix set of LWR-Pressure Vessel (PV) Surveillance Dosimetry Improvement Program (SDIP) standards. In this regard, he has provided overall coordination and has contributed to the preparation of a very large series of U.S. Nuclear Regulatory Commission (NRC) LWR-PV-SDIP progress and technical reports that provide the physics-dosimetry-metallurgy reference documentation that is needed to support the development, use and application of ASTM standards for PWR & BWR operations and surveillance programs. This overall coordination was by NRC contract with Chuck Serpan, Al Taboada, Neil Randall, and Lambros Lois under the auspices of the Offices of Nuclear Regulatory Research & Nuclear Reactor Regulation of the NRC.

He is the lead instructor and author for the ASTM Committee E10 on Nuclear Technology and Applications One-day Standards Technology Training (STT) Course on "Condition Assessment and Surveillance of Nuclear Reactor Pressure Vessel Steels;" this course has been presented world wide. The other instructors and authors are: E. P. Lippincott (Retired from Westinghouse Electric Corporation), A. L. Lowe, Jr. (Retired from Babcock & Wilcox Company), and the late P. D. Hedgecock (NUTECH Engineers, Inc. and Past Chairman of ASTM Subcommittee E10.02 on Behavior and Use of Nuclear Structural Materials). A Second Edition of the One-day ASTM STT Course "Workbook" was completed and issued in November 1991. The authors state:

> *"This One-day Workshop provides an overview of the basic principles, techniques, and methodology to be used in the analysis and interpretation of neutron exposure data obtained from light water reactor pressure vessel surveillance programs; and, based on the results of that analysis, establishes a formalism to be used to evaluate the present and future condition of the pressure vessel and its support structures.*

AEA Reactor Services, United Kingdom, in association with ASTM, organized a UK Workshop on "Radiation Damage Correlation Methodology" that was held at the ASTM European Office, Hitchin, Herts, England in December 1991. ASTM presented its One-day STT Course Workshop that provided a broad based introduction to the subject of pressure vessel condition assessment and surveillance which was a central issue of the AEA-UK Workshop.

This provided an opportunity for the ASTM instructors, UK-AEA Technology staff, UK-HM Nuclear Installations Inspectorate (NII) staff, UK-Nuclear Electric (NE) plc staff, and Scottish Nuclear plc staff to exchange views. Particular emphasis was placed on the exchange and discussion of the most current physics-dosimetry-metallurgy information that has been (or will be) recommended for use in existing codes, regulatory guides, and standards for the management of age and radiation related degradation of materials and reactor components.

In this regard, in depth consideration was given to the status of state-of-the-art development work on damage correlation methodology and its applicability to NE's and NII's responsibilities, respectively, to safely manage and regulate the operation of the UK Sizewell "B" PWR and to extend the operating life of UK Magnox gas cooled nuclear power plants.

Dr. McElroy is the lead instructor and author for another ASTM Committee E10 on Nuclear Technology and Applications One-day STT Course on "Radiological Decontamination & Decommissioning;" this course has only been presented within the U.S in conjunction with ASTM Committee E10 meetings. The other instructors and authors are: P. E. Fuller, (Consulting Services, New Hartford; Past Staff Member of American Nuclear Insurers; Secretary of E10.03), Raymond Gold (Metrology Control Corporation), J. L. Helm (Columbia University), A. S. Kumar (University of Missouri-Rolla), E. P. Lippincott (Consulting Services, Monroeville), R. H. Meservey (EG&G Idaho Incorporated, Chairman of E10.03 on D&D & Extended Life Operation of Nuclear Facilities), P. S. Olson (Rockwell International, Past Chairman of Committee E10), and G. Subbaraman (Rockwell International). A Second Edition of the One-day ASTM STT Course "Workbook" was completed and issued in June, 1995. The authors state:

> "This One-day Workshop provides an overview of

- *Radiological Decontamination and Decommissioning (D&D) and the associated Demolition for nuclear facilities; including research, test, and power reactors, fuel processing facilities, and uranium mills and mining operations.*
- *Status of development and application of a broad range of ASTM standards and supporting documentation that have and are being developed for Nuclear Environmental Restoration and Waste Management (EM) as well as Environmental Assessment and Risk Management (EARM).*
- *Building consensus through risk assessment management of DOE EM program: That is, thru 1) systematic, integrated risk assessment with full stakeholder participation, 2) progress in science-based risk assessment and management, and 3) President's Commission on Risk Assessment & Management & Consortium for Risk Evaluation & Stakeholder Participation (CRESP).*

- *Importance of detailed planning, guidance and implementation of well documented and standardized procedures and Quality Assurance (QA) practices to minimize risks & costs and commentary on the political and technical consensus processes.*
- *D&D concerns over environmental related technological, regulatory, codes & standards, political, insurance, underwriting, claims, legal, economic, and health & safety issues [including those associated with the Department of Labor's Occupational Safety & Health Administration's (OSHA) regulations bearing on Demolition].*
- *Nuclear site specific issues as they are related to current ASTM standards work.".*

Dr. McElroy's expertise has been significantly enhanced by his direct involvement in the development and presentation of the two ASTM STT One-day Course Workshops. Of particular value has been the interactions and exchange of the most current views and information with course students from nuclear related industries, utilities, universities, and regulatory bodies in the U.S. and other countries.

Dr McElroy's career has included positions as: Senior Research Eng'r at Atomics International (Rockwell International) in the design, development, testing, operation and licensing of research reactors; Manager of Reactor Operations at the IIT Research Institute; Westinghouse Hanford Co. (WHC) Fellow Scientist and Manager of the Irradiation Analysis Section at the Hanford Engineering Development Laboratory (HEDL); Manager of the National Dosimetry Center (NDC) at WHC-HEDL. As a part of his nuclear reactor work experience, he has been a reactor operations supervisor, prepared reactor operation and maintenance manuals and provided direction, training and examinations for reactor operations personnel.

He is retired from WHC and Battelle, Pacific Northwest Laboratories (PNL), where he was the Manager of the WHC and PNL National Dosimetry Centers (NDC). He currently is the President of Consultants and Technology Services (CTS), has a B.S. in Chemistry from the University of Southern California and a M.S and Ph.D in Physics from the Illinois Institute of Technology (IIT).

ACRONYMS

AMES	Ageing Materials Evaluation and Studies
AEA	Atomic Energy Authority, United Kingdom
AMP-II	A Modular Code System for Generating Coupled Multigroup Neutron Gamma Libraries for ENDF Format, ORNL
ANISN	One-Dimensional Discrete Ordinates Transport Code, ORNL
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	Boiler & Pressure Vessel Code
B&W	Babcox & Wilcox
BWR	Boiling Water Reactor
BCL	Battelle Columbus Laboratories
BMI	Battelle Memorial Institute
BUGLE-96	A Revised Multi-Group Cross Section Library for LWR Applications Based on ENDF/B-VI Release 3
CASMO-3	A Fuel Assembly Burnup Program, Studsvik Energitehnik
CE	Combustion Engineering
CE	Consumers Energy
CEN/SCK	Center for the Study of Nuclear Energy, Mol, Belgium
CF	Chemistry Factor
CFR	Code of Federal Regulation
COSA2	Spectrum Adjustment System Based on the STAY'SL Type Least Squares Code, Rossendorf, Germany
CSEWG	Cross Section Evaluation Working Group, BNL
CTS	Consultants and Technology Services, Richland Washington, U.S.A
DPA	Displacement per Atom, an Irradiation Exposure Parameter
DOE	U.S. Department of Energy
DOT-4	One & Two-Dimens'al Neutron/Photon Discrete Ordinates Transport Code, ORNL
DORT	Two-Dimensional Discrete Ordinates Transport Code, ORNL
DOMPAC	Triton Reactor Thermal Shield and Pressure Vessel Mockup, France
DOSCROS84	Dosimetry Cross Section Library in 640 Groups of SAND II Type, Netherlands
DOTSOR	A Module in the LEPRICON Computer Code System for Representing the Neutron Distribution in LWR Cores
DOTSYN	A Module in the LEPRICON Computer Code System for Synthesizing Three-Dimensional Fluxes
ECC	European Community Commission
ECN	Netherlands Energy Research Foundation
EDB	Embrittlement Data Base
EFPY	Effective Full-Power Years
ENDF	Evaluated Nuclear Data File
EOL	End-of-Life
EPRI	Electric Power Research Institute
EURLIB	120 Group Coupled Neutron & Gamma Data Library, ECC EURATOM, CCR-Ispra
ELXSIR	Cross Section Library for LWR Pressure Vessel Irradiation Studies: Part of LEPRICON Computer Code System

EURATOM	Commission of the European Communities
EWGRD	European Working Group on Reactor Dosimetry
FERRET -SANDII	Least-Squares Adjustment Code System Used by HEDL/PNL-NDC & W
FMIPA	Freely Migrating Interstials Per Atom
FMVPA	Freely Migrating Vacancies Per Atom
FORSS	A Sensitivity and Uncertainty Analysis Code System, ORNL
FSAR	Final Safety Analysis Review
GE	General Electric
HAFM	Helium Accumulation Fluence Monitor
HEDL	Hanford Engineering Development Laboratory
HFIR	High Flux Isotope Reactor at ORNL
IAEA	International Atomic Energy Agency
ILRR	Interlaboratory LMFBR Reaction Rate Program
IRDF 90	International Reactor Dosimetry File, IAEA
JAERI	Japanese Atomic Energy Research Institute
JDF-1.1	JENDL Dosimetry File-1.1
JENDL	Japanese Dosimetry File for Material Dosimetry in the JOYO Reactor
JOYO	Experimental Fast Reactor at Oarai Engineering Center of PNC
KORPAS	Irradiation Facility at the RBT-6 Reactor in Russia for Reactor Pressure Vessel Steel Specimen Irradiations and Dosimetry Benchmarking: See PSF Description
LANL	Los Alamos National Laboratory
LEPRICON	Generalized Linear Least-Squares Combination Procedure Code System for PWR Pressure Vessel Surveillance Dosimetry Analysis, ORNL
LSL-M2	Statistical, Least Squares, Log Normal A Priori & Posteriori Distrib'n Code, ORNL
LMFBR	Liquid Metal Fast Breeder Reactor
LR-O	VVER Type Reactor Benchmark
LWR	Light Water Reactor
LWR-PV-SDIP	LWR Pressure Vessel Surveillance Dosimetry Improvement Program
MCBEND	Monte Carlo Code, UK
MCC	Metrology Control Corporation, Richland, Washington, U.S.A
MCNP	A General Monte Carlo N-Particle Transport Code, LANL
MDRF	Materials Dosimetry Reference Facility at University of Michigan: Developed by NIST and University of Michigan
MVP/GMVP	General Purpose Monte Carlo Codes for Neutron and Photon Transport Calculations Based on continuous Energy and Multigroup Methods, Japan
NDC	National Dosimetry Center
NEI	Nuclear Energy Institute
NESDIP	NESTOR Shielding and Dosimetry Improvement Programme, UK
NESTOR	Source Reactor at Winfrith
NEUPAC(J1)	Statistical, Linear Estimation Adjustment Code, Japan
NIST	National Institute of Standards and Technology, U.S.A
NJOY	Data Processing System (June 1983 Version) Based on the Bondarenko Self-Shielding Factor Approach for Resonance Absorption.
NMF-90	Neutron Metrology File:Integrated Database for Neutron Adjust. Calculations
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRI	Nuclear Research Institute, Rez plc. Czech Republic

NUMARC	Nuclear Management and Resources Council
NUPLEX	Nuclear Utility Plant Life Extension
OECD/NEA	Organization for Economic Cooperation & Development/Nuclear Energy Agency
OMB	Office of Management & Budget, U.S.A
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Research Reactor at ORNL
PCA	Poolside Critical Assembly at the Bulk Shielding Reactor at ORNL
PCA Replica	In the ASPIS Shielding Facility of the NESTOR Low-Flux Experimental Reactor, Winfrith, UK
PLEX	Plant Life Extension
PNL	Pacific Northwest Laboratory, Battelle Memorial Institute
PNP	Palisades Nuclear Plant
PREVIEW	Kernel Based Prog. for Out-of-Core Neut. Fluen'e Est's in VVER-440 Reactors
PSF	Pool Side Facility of the Oak Ridge Research Reactor at ORNL
PSU-DOTSOR	A Revised Form of DOTSOR, Penn State
P-T	Pressure-Temperature
PTS	Pressurized Thermal Shock
PV	Pressure Vessel
PWR	Pressurized Water Reactor
RADAK	Statistical, Linear Estimation Experimental Data Processing Program, UK
RG	NRC Regulatory Guide
RIAR	KORPUS was Created at the RBT-6 Reactor (RIAR) for Reactor Pressure Vessel Steel Specimen and Dosimetry Measurement Irradiations; Russian Physics-Dosimetry-Metallurgy Test Reactor
RM	Radiometric Monitor
RNDC	Russian Nuclear Data Center, Obninsk
RPV	Reactor Pressure Vessel
RT_{NDT}	Reference Nil Ductility Temperature
SAILOR	A Coupled Self-Shielded 47 Neutron 20 Gamma Ray Cross Section Library for Light Water Reactors, ORNL
SAND II	Semi-Iterative Adjustment Code; Interpreted as Application of Least Squares Principle Although Statistical Assumptions are not Spelled Out Explicitly
SCK/CEN	Center for the Study of Nuclear Energy, Mol, Belgium
SDO	Standards Development Organization
SIMULATE-3	Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik of America
SENSAK	Statistical, Linear Estimation Adjustment Code, UK
SINBAD	A Shielding Integral Benchmark Archive and Database for PC's
SBI RMI	Dosimetry Cross Section Library, ORNL RSIC Data Library Collection
SPECTRA	Statistical, Linear Estimation Adjustment Code, SANDIA
SRM	Standard Reference Material
SSC	Simulated Surveillance Capsule
STAY'SL	Statistical, Linear Estimation Adjustment Code, ORNL
STDTT	Standards Technology Development, Transfer & Training
SSTR	Solid State Track Recorder
SUSD	Cross Section Sensitivity and Uncertainty Package and Extension to 3D Analysis
TG	Task Group

TWODANT	A Code Package for Two-Dimensional Diffusion Accelerated Neutral Particle Transport Code, LANL
THREEDANT	A Code Package for Three-Dimensional Diffusion Accelerated Neutral Particle Transport Code, LANL
TLD	Thermoluminescence Dosimeter
TM	Temperature Monitor
TORT	Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code, ORNL
TRAMO	A Flexible Multigroup Neutron Transport Code Based on the Monte Carlo Method, Germany
TRIPOLI-3	Three Dimensional Monte Carlo Code, France
TWODANT	TWO-Dimensional Diffusion Accelerated Neutral Particle Transport Code, LANL
UK	United Kingdom
U.S.	United States
U.S.A	United States of America
USAEC	United States Atomic Energy Commission
USE	Upper Shelf Energy
USNRC	U. S. Nuclear Regulatory Commission
VB	PSF Void Box Experiment at ORNL
VENUS	Pressure Vessel Mockup at Mol, Belgium
VITAMIN-C and VITAMIN-B	Fine Group Neutron/Photon Cross Section Libraries Derived from ENDF/B-VI Nuclear Data with 171 Energy Groups
VVER	Soviet Designed PWR, same designation as WWER
W	Westinghouse
WGRD-VVER	Working Group on Reactor Dosimetry for VVER Reactors
WHC	Westinghouse Hanford Company
W-NTD	Westinghouse - Nuclear Technology Division
WWER	Soviet Designed PWR, same designation as VVER

1.0 OVERVIEW

REVIEW PROGRAM AND RESULT -- Consumers Energy asked Consultants & Technology Services (**CTS**) to **1)** provide the consulting services of Dr. W. N. McElroy to assist the Owner in performing a review of the Owner's Palisades Nuclear Plant Pressurized Thermal Shock (PTS) submittals to the NRC dated April 4 and 19, 1996 and June 26, 1997 and the NRC December 20, 1996 safety analysis report (SER) and **2)** upon completing the review submit a letter/report to the owner's representative.

Based on the extent of the reviewers knowledge, CTS was to provide answers to the following four questions:

***QUESTION* 1:** Is the Consumers Energy Method (Westinghouse Method) of determining the palisades nuclear plant's best estimate fluence by combining transport calculation and dosimetry measurements technically sound?

***QUESTION* 2:** Is the method used consistent with the basis of 10 CFR 50.61, the PTS Rule?

***QUESTION 3*:** Has Consumers Energy's Explanation of the Bias Between Measurement and Calculations Provided Sufficient Basis to Support the Magnitude of This Difference?

***QUESTION 4*:** Has the Owner Communicated Its Position Clearly and If Not Where Would Further Explanation be Useful?

Further, the consultant was to include any additional items that were felt to be relevant to the discussion between the NRC and Palisades Plant staff.

Based on a very careful review and study of all of the relevant documentation and issues, the answers for Questions 1, 2, 3, and 4 are very definitely yes!

Detailed commentary and supporting documentation related to the above questions & answers as well as the consideration of additional issues are provided in Sections 3 through 9 and the Enclosures/Attachments and Reference Listing.

REVIEW HIGHLIGHTS AND WORK SCHEDULE -- The April 4, 1996 letter to the US Nuclear Regulatory Commission [Sm96] states:

"This letter provides recent reevaluations of Palisades fluence data. New calculated and best estimate fluence values are presented and discussed. The new best estimate fluence values are derived utilizing a bias factor which is based on Palisades in-vessel and ex-vessel capsule measurements. The previously existing capsule measurements have been reevaluated and updated values are provided. Using the information provided in this letter, a revised PTS screening criteria date, as defined in 10CFR50.61, has been calculated. The axial welds containing heat # W5214 remain the limiting vessel material and are now estimated to reach the screening date in the year 2011." ....

Attachment 1 contains the Palisades evaluation of the updated reactor vessel fluence values and capsule fluence measurements. Attachment 2 contains Westinghouse Report WCAP14557, Revision 1, "Consumers Power Company Reactor Vessel Neutron Fluence Measurement Program for Palisades Nuclear Plant - Cycle 1 through 11." Attachment 3 contains AEA Report AEAT-0121, "Fluence Calculations for the Palisades Plant."

NRC staff concurrence with the appropriateness of this updated reactor vessel fluence methodology to project reactor vessel life is requested by November 1, 1996."

A careful review and study of Attachments 1, 2, and 3 was completed by November 8, 1997. At the same time an in depth review and study of other relevant and supporting documentation, as identified in Section 2.0, was initiated to address specific issues that had been raised in the three attachments.

The June 26, 1997 letter to the US Nuclear Regulatory Commission [Bo96] states:

"The NRC Interim Safety Evaluation dated December 20, 1996 concluded that Consumers Energy had determined the Palisades reactor vessel fluence in accordance with Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence." This draft guide requires that plants qualify their calculations methods using available measurements and allows a plant to adjust the calculations using a correction factor determined from a statistically significant measurement data base." ....

With the additional information provided by this letter, we believe that Consumers Energy has provided the staff with a sound technical position to support the additional 17% reduction in vessel fluence recommended by our April 4, 1996 submittal. We believe this technical justification satisfies all Best Estimate analysis requirements and basis of 10 CFR 50.61 and RG 1.99 [Re88]. We therefore, request that the staff approve this fluence reduction as proposed. If the staff is unable to concur with our technical justification for the fluence reduction, we request that the technical basis for the staff position be provided to Consumers Energy as soon as possible."

The June 26, 1997 letter then listed three attachments and six enclosures which provide Consumers Energy's response to issues raised by the NRC staff and information previously placed on the docket are enclosed to facilitate staff review.

A careful review and study of Attachments 1, 2, and 3 and Enclosures 1- 6 was completed by November 13, 1997. At the same time an in depth review and study of other relevant and supporting documentation, as identified in Section 2.0, was continued to address specific issues that had been raised in the three attachments and six enclosures.

The December 20, 1996 NRC letter to Thomas C. Bordine, Manager of Licensing, Palisades Plant [Bo96] states:

"By letter dated April 4, 1996, you submitted an updated neutron fluence evaluation for the Palisades reactor pressure vessel and requested that it be reviewed and accepted as the basis for the 10 CFR 50.61 PTS vessel embrittlement evaluation. You provided additional information in response to staff questions on June 12, June 21, August 27, September 9,

September 19, and October 1, 1996. The submittal claimed a 25 % reduction in reactor vessel fluence from the value accepted in the previous review of the Palisades reactor vessel."

The staff received contract assistance in this review from BNL. The NIST also provided contract assistance for specific dosimetry questions. Our evaluation indicates that an 8% reduction in the calculated fluence due to physical changes in the plant is acceptable. The remaining 17%, consisting of a bias resulting from the comparison of calculated to measured fluence values and a reduction due to spectral adjustments can not be approved at this time. The current Palisades fluence value results in the Palisades reactor vessel reaching the PTS screening criteria in 1999. The 8% reduction approved by this evaluation results in the vessel reaching the screening criteria in 2003.

This is an interim report. Our review effort will continue at BNL, focusing on the acceptability of the calculation-to-measurement bias and the statistical treatment of the data. Our safety evaluation is provided as Enclosure 1. The BNL technical evaluation report is provided as Enclosure 2."

A careful review and study of Enclosures 1 and 2 was completed by November 20, 1997. At the same time an in depth review and study of other relevant and supporting documentation, as identified in Section 2.0, was continued to address specific issues that had been raised in the two enclosures.

The CTS Lette./Report preparation started on November 20, 1997 and was completed on November 26, 1997. This effort continued to involve the in depth review and study of all of the relevant documentation identified in Section 2.0 and preparing written technical responses and justification for answering "yes" to all of the four Questions in Section 2.0. The more difficult task was to provide Consumers Energy with sufficient additional technical documentation and justification so that the NRC would accept the results and allow the additional 17% reduction in reactor vessel fluence for the Palisades Nuclear Plant.

This was accomplished by an in depth technical review and study of the technical basis and soundness of the NRC recommended procedures and methodology found in the Code of Federal Regulations 10 CFR Part 50, Appendix G and H, the PTS Rule 10CFR50.61, RG DG-1025, RG DG-1053, RG 1.99 Rev. 2, ASTM E706 Master Matrix Set of Physics-Dosimety-Metallurgy Surveillance Standards and the ANS 19.10 Standard, "Fast Neutron Fluence to PWR Reactor Cavities;" see Section 3.0, Subsection "PWR & BWR Surveillance Program Regulatory Instruments.

More specifically, it was stated above that *"The NRC Interim Safety Evaluation dated December 20, 1996 concluded that Consumers Energy had determined the Palisades reactor vessel fluence in accordance with Draft Regulatory Guide DG-1053."*

The determination of the Palisades reactor vessel fluence is not based on the RG DG-1053 procedures and methodology, but rather on the application of the Westinghouse Methodology. This Methodology is based on the requirements of the Code of Federal Regulations 10CFR Part 50, Appendix G and H, the PTS Rule 10CFR50.61, Reg. Guide 1.99, Rev. 2, and the referenced ASTM E706 Master Matrix Set of Physics-Dosimetry-Metallurgy Surveillance Standards (See Table 3.1).

As such, the NRC ***needs to*** accept the results generated by the use of the Westinghouse methodology since it is based on the procedures, methodology, and data bases specified in the Code of Federal Regulations 10 CFR Part 50, Appendix G and H, the PTS Rule 10CFR50.61, RG 1.99 Rev. 2, and the referenced ASTM Standards.

Based on the present CTS review and study of the submittals by Consumers Energy's staff and the NRC's staff responses, the NRC has not provided an acceptable technical basis for not giving its concurrence with the appropriateness of the PNP updated reactor vessel fluence methodology to project reactor vessel life.

If the NRC's staff wants specific concerns about the validity of the procedures and methodology recommended in the ASTM standards addressed, this can be accomplished through the ASTM voluntary technical consensus process. That is, by working directly with the task groups that are responsible for reviewing technical issues and making changes through the technical consensus balloting process.

More will be said about this in Section 3.0 since it has to do with Consumers Energy's request for additional items relevant to discussions between NRC and Palisades Plant's staff.

In [Ca96], Carew and Aronson state in **Section VI.1, Best-Estimate Fluence Determination**:

"...............Based on these measurement-to-calculation (M/C) comparisons of the dosimeter reaction rates, a M/C bias of 12(± 7)% is determined. This M/C bias is then adjusted using a least-squares adjustment technique to account for uncertainties in the measurement and calculations. In the case of Palisades this adjustment increases the M/C bias from 12% to 17%, and implies the calculations are over predicting the fluence by 17%. The determination of the M/C bias and the adjustment method are discussed in the following sections."

In **Section VI.2, Fluence Calculation-to-Measurement Bias**, they state:

"From Figure 4, it is seen that the in-vessel M/C bias is 1.00 ± 0.03 for the dosimeters with thresholds E> 4.0 MeV, and 0.86 ± 0.02 for the dosimeters with thresholds E< 4.0 MeV. In the CPC/W analysis this difference is assumed to be due to a spectrum-dependent error in the DORT calculations which results in an exact calculation above E> 4.0 and an over prediction for E< 4.0 MeV. Based on this assumption, a 12% M/C fluence reduction (i.e., not including the additional FERRET reduction) is applied to the DORT E> 1.0 MeV fluence prediction. The application of this M/C spectrum-dependent correction is illustrated in Figure 6. While this conclusion may be correct, there are several other possible explanations for the observed 1.00/0.86 difference between the high/low-energy M/C biases that would not require this reduction in the DORT calculated fluence. These include: 1) the use of erroneously low dosimeter cross sections for Fe-54 and Ni-58 in the interpretation of the measurements and/or 2) errors in the Fe-54 and Ni-58 measurements. (The number of U-238 and NP-237 dosimeters that are included in the in-vessel M/C bias is small and these measurements are subject to relatively large uncertainties.)"

In **Section VI.3, Least-Squares Fluence Adjustment**, they state:

"A major concern with the application of the FERRET adjustment is that, while the adjustment

does provide a best-fit of the measured data, the dosimeter cross sections, measured reaction rates and calculated spectrum adjustments are made without any physical basis. This application of the FERRET adjustment methodology to Palisades is presently being evaluated and the results of this evaluation will be reported separately when completed."

That part of the above statement, "the dosimeter cross sections, measured reaction rates and calculated spectrum adjustments are made without any physical basis," is not a correct technical statement and it contradicts what is scientifically known and has been accepted (via the established world wide ASTM voluntary technical consensus process) in the applicable ASTM E706 Master Matrix Set of LWR Physics-Dosimetry-Metallurgy Surveillance Standards; see Section 9.

All of the above three statements have to do with questions and answers about the validity of the procedures and methodology recommended in DG 1053 and the referenced ASTM and ANS 19.10 standards. Obviously, such questions must be addressed and any needed clarification should be provided in DG 1053 and the referenced ASTM and ANS 19.10 standards to add more certainty (and cost effectiveness) to the process of the regulation and safe operation of nuclear power plants.

How to "**Add More Certainty**" to the regulatory process through Standards Technology Development, Transfer and Training (**STDTT**) and the discussion of "**Key Issues**" are considered in Section 3.0, Subsection **FLR & STDTT**.

An important aspect of this letter/report is to clarify some of the misunderstandings that Lois, Carew, Aronson and others may have developed as a result of Worsham's two technical papers [Wo96a,Wo96b] that were distributed at the 9th ASTM-EWGRD Symposium on Reactor Dosimetry, Prague, Czech Republic, September 2-6, 1996; copies of these two papers are attached. By the incorrect interpretation and use of the Pool Critical Assembly (**PCA**) [1] reported FERRET-SAND II and LSL-MS2 least-squares adjustment code results, Worsham concluded that [Wo96b]:

*"When FTI began evaluation of the calculational requirements of the new draft guide, a clear implication was that surveillance capsule fluence calculations must be within the uncertainty of previous measurements. The "Margin" term associated with **a)** RG 1.99, Rev. 2, and **b)** the generic PTS safety analysis, requires that the uncertainty in the capsule calculational predictions be less than or equal the uncertainty in the previous measurement predictions. Unfortunately, FTI found that the previous fluence "measurement" predictions in the RG 1.99 embrittlement data base are biased. Investigations of a) the work of Simons, et alia, Lippincott, et alia, and Stallmann, et alia, and b) the reasons for the biased measurements, suggest that the biases are caused by the old FERRET-SAND technology."*

It is noted that Lois attended the 9th ASTM-EWGRD Symposium and very likely talked to

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[1] PWR-PV Mockup Experiments and Calculational Blind Test.

Worsham and received copies of his papers. [2] Worsham concluded that least-squares adjustment codes should not be used for obtaining the best-estimate value of the PV fluence. This issue is given further consideration in Section 3.0 under the Subsection "Biased Fluences in the Charpy Embrittlement Database."

His two 9th ASTM-EUROPEAN papers [Wo86a,Wo86b] were carefully reviewed and studied to see if there were technical merit for the conclusions presented in his two papers. A technical justification could not be found as reported in Section 3.0, Subsection "Biased Fluence in the Charpy Emrittlement Database."

Further, in the Minutes for the June 2, 1997 Orlando ANS-19.10 Meeting [Lo97], it states:

"T. Worsham stated that he was concerned that the fluences in the Charpy embrittlement database may be biased. He noted that the use of LEPRICON is an interesting concept, but is concerned that these codes do not have a physical basis for the adjustment. In addition, he suggested that any fluence adjustment be made directly to the transport calculations rather than through FERRET and LEPRICON."

The International-Interlaboratory PSF Physics-Dosimetry-Metallurgy Experiment yielded the highest quality and accurate physics-dosimetry results reported for a PWR Pressure Vessel Benchmark Mockup Experiment [Mc87b]. The $^{54}Fe(n,p)$, $^{58}Ni(n,p)$, $^{56}Ti(n,p)$, $^{63}Cu(n,\alpha)$, $^{238}U(n,f)$, and $^{237}Np(n,f)$ and other sensors used in this experiment are those recommended (in the ASTM standards) for use in PWR and BWR surveillance capsules to measure the flux and fluence E> 1 MeV. These are the same sensors used for the PSF physics-dosimetry-metallurgy experiments and in the Palisades surveillance capsules. The technical issues associated with the application of the ASTM Master Matrix Standards and FERRET-SAND II Adjustment Code Methodology for determining the PNP "Best Estimate" of the fluence and answers to specific technical questions raised by NRC staff are provided in Sections 3 & 9.

Detailed supporting documentation is presented in Section 3.0, Subsection **KEY ISSUES**, that provides necessary justification to refute the fact that there is a need for having a "physical basis" to justify the use of a least-squares adjustment code for generating the most accurate "Best Estimate" fluence value for the determination of the PTS Rule pressure vessel embrittlement. Section 5.2 of the "ASTM E944 Standard Guide for Application of Neutron Adjustment Methods in Reactor Surveillance" specifies the "Requirements for the Use of Adjustment Codes in Reactor Surveillance." Additional commentary is provided in Table 3.1, PWR & BWR Surveillance Program Regulatory Instruments, and the Table 3.1 Attachment.

In view of FTI continued support of Worsham in assisting in the preparation and revision of the draft of the new ANS-19.10 "Fast Neutron Fluence to PWR Reactor Cavities" standard, Consumers Energy should encourage its staff to continue (along with Stan Anderson) its active participation in the standards development work for this new LWR physics-dosimetry

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[2] It is noted that the peer review of the papers and subsequent publication of the proceedings of the 9th Symposium is still in progress; as such, only the attendees would have received copies of Worsham's two papers.

surveillance standard. Anderson has already prepared a draft of Section 5.0 of this standard on "Determination of Best Estimate Fluence;" see Enclosure 3. This draft is very well written and uses appropriate information extracted from DG 1053, ASTM Standard E706-(IIE2), "Guide for Benchmark Testing of LWR Calculations," and ASTM Standard E944 on "Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance." It would be very appropriate to use this write-up for inclusion in a future revision of DG 1053.

2.0 INTRODUCTION

Based on the extent of the reviewers knowledge, provide answers to the following questions:

QUESTION 1: Is the Consumers Energy Method (Westinghouse Method) of determining the palisades nuclear plant's best estimate fluence by combining transport calculation and dosimetry measurements technically sound?

QUESTION 2: Is the method used consistent with the basis of 10 CFR 50.61, the PTS Rule?

QUESTION 3: Has Consumers Energy's Explanation of the Bias Between Measurement and Calculations Provided Sufficient Basis to Support the Magnitude of This Difference?

QUESTION 4: Has the Owner Communicated Its Position Clearly and If Not Where Would Further Explanation be Useful?

In order to provide answers for these four questions it was necessary to:

1) Review and study the following relevant documentation [See Reference Listing]:

- C. Serpan's Prior Papers Related to NRC's Research & Regulatory Positions on the Physics-Dosimetry Derivation of the PTS Rule's Fluence Estimate.
- A. Taboada's Prior Papers Related to NRC's Research & Regulatory Positions on the Physics-Dosimetry Derivation of the PTS Rule's Fluence Estimate.
- N. Randall's Prior Papers Related to NRC's Research & Regulatory Positions on the Physics-Dosimetry Derivation of the PTS Rule's Fluence Estimate.
- L. Lois's Prior Papers Related to NRC's Research & Regulatory Positions on the Physics-Dosimetry Derivation of the PTS Rule's Fluence Estimate.
- J. Carew's Prior Papers Related to NRC's Research & Regulatory Positions on the Physics-Dosimetry Derivation of the PTS Rule's Fluence Estimate.
- The New 1996 Federal Public Law #104-113 on the "Technology Transfer Improvements Act (**TTIA**) of 1995," Code of Federal Regulations 10 CFR Part 50, Appendix G & Appendix H, PTS Rule 10CFR50.61, RG DG-1025, RG DG-1053, RG 1.99 Rev. 2, and the ANS 19.10 Standard, "Fast Neutron Fluence to PWR Reactor Cavities." [Lo97]
- Relevant NRC LWR-PV-SDIP Physics-Dosimetry-Metallurgy NUREG/CR-XX Program Reports.
- Relevant ASTM Physics-Dosimetry Related Standards and ASTM Subcommittees E10.02 and E10.05 Meeting Minutes and W. N. McElroy's March 30, 1996 ASTM Task Group E10.05.04 Report on the E706 Master Matrix Set of LWR Surveillance Standards: See Attached copy.
- Relevant Prior ASTM-EURATOM International Symposia on Reactor Dosimetry

Papers, with Emphasis on those Presented at the 9th Symposium, September 2-6, 1996, Prague, Czech Republic. This Symposium was Attended by Lambros Lois as a Result, at Least Partially, of Discussions with W. N. McElroy and his Sending Lambros a Copy of the March 30, 1996 E10.05.04 TG Report Identifying Those World Wide Assisting in Revision, Drafting and/or Review of the ASTM E706 Physics-Dosimetry-Metallurgy Set of LWR Surveillance Standards.

2) Review and study the Owner's Palisades Nuclear Plant's (**PNP**) Information Package on Pressurized Thermal Shock (**PTS**) submittals to the NRC dated April 4, 1996 and June 26, 1997 and the NRC's December 20,1996 Interim Safety Evaluation (**ISE**).

3) Provide information on any additional ***"ISSUES"*** that are relevant to the discussions between the NRC and the Palisades Plant Staff.

Based on a very careful review and study of all of the relevant documentation and issues, the answers for Questions 1, 2, 3, and 4 are very definitely yes!

Commentary and supporting documentation related to the above questions & answers as well as the consideration of additional issues are provided in Sections 3 through 9 and the Enclosures/Attachments and Reference Listing.

3.0 FEDERAL LAW & REGULATIONS (FLR); STANDARDS TECHNOLOGY DEVELOPMENT, TRANSFER & TRAINING (STDTT); AND KEY ISSUES AND SUPPORTING TECHNICAL DOCUMENTATION THAT ARE RELEVANT TO FLR & STDTT AND PAST & FUTURE DISCUSSIONS BETWEEN THE NRC AND PALISADES NUCLEAR PLANT (PNP) STAFF

FLR & STDTT -- In order to conclude that the answer was yes to the four questions in Section 2, it was necessary to review and study the new 1996 Federal Public Law 104-113 on the "Technology Transfer Improvements Act (**TTIA**) of 1995, the Code of Federal Regulations 10 CFR Part 50, Appendix G & H, the PTS Rule 10CFR50.61, RG DG-1025, RG DG-1053, RG 1.99 Rev. 2, the referenced ASTM Master Matrix Set of LWR Physics-Dosimetry-Metallurgy Surveillance Standards, and the ANS 19.10 Standard, "Fast Neutron Fluence to PWR Reactor Cavities."

Contained in the new TTIA law is a provision 12(d) that codifies the existing Office of Management and Budget (**OMB**) Circular A-119 on Federal Participation in the Development and Use of Voluntary Standards. The law directs that all Federal agencies and departments shall use ***"Technical Standards"*** that are developed or adopted by voluntary consensus standards bodies, using such technical standards as a means to carry out policy objectives or activities determined by the agencies and departments. The National Research Council of the Academy of Science & Engineering report related to the new law says that *"voluntary consensus standards are often equally as stringent in the level of protection they require as mandatory standards would be. It might seem reasonable to expect that private standards developers--industry associations, especially--would seek to set standards at the lowest common denominator of safety. .....In fact, however, private standards writers have several incentives to set high standards. Forestalling government regulation by developing a private solution to a perceived problem requires a standard stringent enough to satisfy public needs. (Government participation in standards committees enhances this process from both public and private perspectives.)..."* [Ko96]

The definition of "Technical Standards" as used in Subsection 12(d) means performance-based or design-specific technical specifications and related management systems practices.

The National Institute of Standards & Technology (**NIST**) has posted an implementation plan for Public Law 104-113 on the Web [http://ts.nist.gov/ts/htdocs/210/plan.htm]. The law directs NIST to lead a national effort to coordinate standards and conformity assessment activities among federal, state and local government agencies, and the private sector.

The Private Sector, ASTM, ASME, ANS, NRC, and DOE should consider meeting with the NIST staff to develop a coordinated national STDTT effort to more effectively specify, manage, and make use of the Code of Federal Regulations 10 CFR Part 50, Appendix G & Appendix H, the PTS Rule 10CFR50.61, RG DG-1053, RG 1.99 Rev. 2, the ANS 19.10 Standard, and the ASTM E706 Master Matrix Set of Physics-Dosimetry-Metallurgy LWR Surveillance Standards. The use and modification of the FTR Dosimetry Handbook [Li83] would provide a good starting point for such a coordinated effort; that is, it could be updated and modified to serve as an important reference source for research, test, and power reactor applications and surveillance programs. A tremendous amount of LMFBR research and development money was used to develop and maintain the HEDL National Dosimetry Center, which is still being maintained, but as the PNL National Dosimetry Center.

In an 8th ASTM-EURATOM Symposium paper on "Reactor Vessel Dosimetry Assessment: Perspective of the Materials Engineer" by Steele et al., it is stated:

> *"Material and dosimetry specialists, who develop and implement surveillance programs and resolve regulatory concerns for individual reactor vessels primarily utilize three ASTM publications: ASTM Standard E853-87, ASTM Standard E1005-84 (Reapproved 1991), and ASTM Standards Technology Training Course and Workbook [**Mc91**]."*
>
> *With the assistance of experts and careful use of all referenced standard guides, practices and related references, the materials specialist and other appropriate reactor plant personnel should be capable of determining the effect of neutron exposure in terms of remaining reactor vessel design and safety margins. A case history dealing with the Yankee Rowe reactor vessel issue is presented next. Prior to looking at this case history, the current ASTM-specified path and potential impediments to resolving outstanding issues are assessed. The pattern of Standard E853-87 sounds straightforward and is summarized as follows: ....*
>
> *There is predominant emphasis on core physics, operating history, and dosimetry computations. In addition, E853-87 refers to 21 other standards and guidelines, has 70 references, and refers to major NRC Code of Federal Regulations and ASME documents. For materials specialists who are not dosimetry specialists, this approach is highly complex, time consuming and difficult to use. ...."*

To the three ASTM publications mentioned above, one must add others from those discussed in the March 30, 1996 E10.05.04 TG Report; such as ASTM Standards E185, E482, E560, E900, and E944 as well as NRC RG 1.99, Rev. 2, the Draft DG 1053 on Calculational and Dosimetry Methods for Determining PV Neutron Fluence, and the ANS 19.10 Standard on Fast Neutron Fluence to PWR Reactor Cavities.

The complexity and difficulty of having to understand, apply, and use all of these different standards and regulatory instruments is very great and challenging for the U.S. nuclear industry. This must be kept clearly in mind for those accepting the responsibility for the development, drafting, and revision of the ASTM E706 Master Matrix Set of LWR Surveillance and ANS 19,10 Standards.

Additional commentary on Codes, Regulations, Regulatory Guides, and how the ASTM E706 'Set of LWR Surveillance Standards are identified through the ASTM E185 Standard as a part of Appendix H to 10 CFR 50 and as such, by reference, are made a part of CFR will be found on pages 2 thru 8 of the March 30, 1996 TG Report,

Lowe's commentary on the "Critical Role of Neutron Fluence in Reactor Vessel Integrity" will be found on pages 7-8 of the March 30th Task Group report, Enclosure 4. He states:

> *In the U.S, the situation is enhanced by the wording of 10CFR50, Appendix G. While there is reference to change in material properties, nowhere in this appendix is there*

any reference to neutron fluence, dosimetry, or analysis procedures. To assess the effects of neutron fluence on the material properties as needed for Appendix G analysis, 10CFR50, Appendix G references 10CFR50, Appendix H. A review of this appendix has minimum reference to the neutronic requirements of the surveillance program except for an integrated surveillance program: "...there must be an adequate dosimetry program for each reactor." The remainder of the requirements are buried in ASTM Standard E185. There is little wonder that the role of dosimetry and fluence analysis in reactor vessel integrity is not appreciated....

The publication of 10CFR50.61......,. was the first regulation that highlighted the role of reactor vessel neutron fluence on operating integrity. This regulation establishes a direct relationship between the reactor vessel screening limitations and the reactor vessel neutron fluence.....

The PTS issue greatly increased the industry's appreciation of the importance of the measurement, accuracy, and precision of fluence calculations.

In summary, reactor vessel integrity is based on a number of properties that must be understood, including nondestructive evaluation of the critical beltline materials, knowledge of effects of irradiation on material properties, and accurate knowledge of neutron fluence. Like a three-legged stool, all three elements are essential to ensure reactor vessel integrity."

With the passage of TTIA, the opportunity now exists for the nuclear industry to bring much more certainty to the process of the federal regulation of the safety of the operation, surveillance programs, and decommissioning programs for nuclear power plants. As examples, additional information is provided in **Attachment I** on how ASTM voluntary technical standards are bringing **much more certainty** to the process of Environmental Assessment and Environmental Risk Management. This information was taken from Reference [Mc95].

PWR & BWR SURVEILLANCE PROGRAM REGULATORY INSTRUMENTS

Table 3.1 provides commentary on PWR and BWR surveillance program regulatory instruments. With this information in hand, it appears that the Regulatory Guide DG 1053 methodology does not recommend and support the determination of the "best estimate" of the EOL pressure vessel fluence value by use of the least-squares method and the adjustment codes recommended in the E944 Guide; i.e., E944 is never mentioned or referenced in DG 1053. As such, the DG 1053 procedures and methodology lack the required acceptance (technical consensus) from the industry and the public, and is, therefore, not appropriate for providing regulatory guidance as it is now written.

That is, this review of the Regulatory Instruments used for the regulation of surveillance programs for PWR and BWR pressure vessels clearly demonstrates that the recommended methodology of the new DG 1053 is not consistent with that recommended in the ASTM E706 Master Matrix Set of LWR Physics-Dosimetry-Metallurgy Surveillance Standards that have been developed and adopted over the last ~ 35 years on a world wide basis by the industry and the public through voluntary technical consensus due process; see Section 9. Since DG 1053 lacks world wide voluntary technical consensus due process, it needs to be revised to be consistent with the procedures and methodology recommended in the ASTM E706 set of standards. Further, the implementation of the use of the DG 1053 as it is now written adds confusion and make it very difficult for licensee's to remain in compliance with Federal Law as now specified in 10 CFR Part 50, Appendix G & H, the PTS Rule 10CFR50.61, Reg. Guide 1.99, Rev. 2, and the ASTM E706 Master Matrix Set of LWR Surveillance Standards (that by reference to the ASTM E185 Standard in Appendix H has made the entire set of E706 standards a part of Federal Law).

In this regard, on pages 65458 & 65459 of the Federal Register / Vol. 60, No. 243 / Tuesday, December 19, 1995 / Rules and Regulations, changes in Appendix H of 10 CFR 50 are discussed. It is stated that:

"The other principal change to Appendix H clarifies the version of ASTM Standard E185 that applies to various portions of the surveillance programs. Appendix H recognizes the need to separate surveillance programs into two essential parts. Specifically the design of the program and the subsequent testing and reporting of results from surveillance capsules. Because the design of the surveillance program cannot be changed once the program is in place, the requirements for design of the surveillance program are static for each plant. However, the testing and reporting requirements are updated along with technical improvements made to ASTM standard E185."

TABLE 3.1

PWR & BWR SURVEILLANCE PROGRAM REGULATORY INSTRUMENTS

- As discussed in [Mc88], for each LWR nuclear power plant, the physics-dosimetry-metallurgy surveillance program requirements are identified through the **ASTM Standard E185 on "Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706(IF),"** which in turn is identified as a part of the 10 CFR PART 50, Appendix H.

- The requirements for design of the surveillance program are static for each plant. However, the testing and reporting requirements are updated along with technical improvements made to the ASTM E185 Standard.

- **Section 8.2.1 of ASTM E185-94 states that:** *"The neutron fluence rate, neutron energy spectrum, and neutron fluence of the surveillance specimens and the corresponding maximum values for the reactor vessel shall be determined in* ***accordance with the guidelines in ASTM Standard E482 on "Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance" and ASTM Standard E560 on "Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results."***

  Section 8.2.2 states that: *"The specific method of determination shall be documented."*

  Section 8.2.3 states that: *"The Neutron fluence rate and fluence values (E>0.1 and 1 MeV) and dpa rate and dpa values shall be determined and recorded using a calculated spectrum adjusted or validated by dosimetry measurements."*

  There is no existing regulatory requirement that the prediction of the vessel fluence must be made by an "**absolute**" fluence calculation in which the transport of the neutrons from the core is calculated out to the vessel cavity, rather than a simple spatial extrapolation of the fluence measurements. This is a new requirement in DG 1053 and it is inconsistent with the existing Federal Law's regulatory requirements of the ASTM E706 Master Matrix Set of Physics-Dosimetry-Metallurgy Standards. Furthermore, this provision has not been considered by ASTM Committee E10 and subjected to ASTM technical consensus due process.

  Draft Regulatory Guide DG 1053 also states that:

  "Compliance with this guide is not a regulatory requirement of the USNRC. However, if a licensee elects to use this guide to determine pressure vessel neutron fluence, implementation of the guide would not be satisfied unless the licensee complies with certain specific provisions identified in the Regulatory Position of the guide."

TABLE 3.1 (Cond't)

- **Section 3.1.1 of ASTM Guide E482 states that:**

  "The methodology recommended in this guide specifies criteria for validating computational methods and outlines procedures applicable to pressure vessel related neutronic calculations for test and power reactors.

- **Section 3.2 Validation states that:**

  Prior to performing transport calculations for a particular facility, the computational methods must be validated by comparing results with measurements made on a benchmark experiment. Criteria for establishing a benchmark for the purpose of validating neutronic methodology should include those set forth in ***ASTM Standard 944 on Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance"*** *as well as those prescribed in* ***Section 3.2.1. Requirements for Benchmarks.***

- **Section 3.4.8.2 states that:**

  Use a spectrum adjustment procedure as recommended in Guide E944 using calculated group fluences and dosimetry data with uncertainty estimates to obtain an adjustment to the calculated group fluences and exposure parameters. Predicted pressure vessel fluences could then incorporate the spectral and normalization data obtained from the adjusted fluences.

- **Section 4.2.3 of ASTM Practice E560 states that:**

  "Guide E944 should be used to combine the transport calculation with the dosimeter results. The E944 adjustment procedure should be used to indicate whether the dosimeter measurements and associated uncertainties are consistent with the transport calculation and with uncertainties implied from benchmark tests of the transport code (PCA, VENUS, NESDIP, and appropriate Commercial BWR or PWR). Having established the required consistency, the adjusted transport code results may be used to calculate the neutron field at all points in the pressure vessel wall with the uncertainties estimates derived from the application of the adjustment codes. Direct use of the transport code results with the appropriate (experimentally determined) bias factors and uncertainties is another acceptable approach."

- **Section 3.1 of ASTM Guide E944 states that:**

  "Adjustment methods provide a means for combining the results of neutron transport calculations with neutron dosimetry measurements in order to obtain optimal estimates of neutron damage exposure parameters with assigned uncertainties. The inclusion of measurements reduces the uncertainties for these parameter values and provides a test

TABLE 3.1 (Cond't)

for the consistency between measurements and calculations and between different measurements. This does not, however, imply that the standards for measurements and calculations of the input data can be lowered; the results of any adjustment procedure can be only as reliable as are the input data."

- **Appendix G of 10 CFR Part 50 specifies the:**

  "fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light-water-cooled nuclear power reactors."

- **Appendix H of 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements"**

  "Provides the rules for monitoring the changes in the fracture toughness properties of the RPV beltline materials due to irradiation embrittlement using a surveillance program. ***Appendix H references ASTM Standard E185*** *for many of the detailed requirements of surveillance programs, and permits the use of integrated surveillance programs, wherein surveillance program capsules for one reactor are irradiated in another reactor."*

- **PTS Rule 10CFR50.61 requirements state that:**

  "For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{PTS} must use the calculation procedures given in paragraph (c)(1) of this section, except as provided in paragraphs (c)(2) and (c)(3) of this section."

  Section (c) states that:

  "Calculation of RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. RT_{PTS} must be evaluated using the same procedures used to calculate RT_{NDT} as indicated in paragraph (c)(1) of this section, and as provided in paragraphs (c)(2) and (c)(3) of this section."

  Section (c)(1)(iv)(B) states that:

  "f is the best estimate neutron fluence, in units of 10^{19} n/cm^2 (E > 1 MeV) at the clad-base metal interface on the inside surface of the vessel at the location where that material in question receives the highest fluence for the period of service in question. As specified in this paragraph, the EOL fluence for the vessel beltline material is used in calculating ΔRT_{PTS}."

TABLE 3.1 (Cond't)

- **Regulatory Guide 1.99, Rev. 2: In Section B, Discussion, it is stated:**

"The basis for Equation 2 for ΔRT_{NDT} is contained in publications by G.L Guthrie [Gu84] and G.R. Odette et al. [Od84].

"The measure of fluence used in this guide is the number of neutrons per square centimeter having energies greater than 1 million electron volts (E>1 MeV). The differences in energy spectra at the surveillance capsule and the vessel inner surface locations do not appear to be great enough to warrant the use of a damage function such as displacements per atom (dpa) in the analysis of the surveillance data base [Mc87a]." (It is noted that Guthrie made use of the reference [Mc87a surveillance capsule derived FERRET SAND II values of fluence while Odette, primarily, use derived fluence values based on only the measured ^{54}Fe reaction rate. The Table 3.1 Attachment provides additional information related to the basis and uncertainties associated with the procedures and methodology of RG 1.99, Rev. 2.)

It is also important to know that **RG 1.99, Rev. 2** is essentially the same as the **ASTM Standard E900 on "Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials."** RG 1.99, Rev. 2, therefore has been developed on the basis of ASTM voluntary technical consensus due process. It is stated in Section 1.3 of E900 that:

"This guide is Part IIF of the Master Matrix E706 which coordinates several standards used for irradiation surveillance of light-water reactor vessel materials. Methods of determining the applicable fluence for use in this guide are addressed in Master Matrix E706, Practices E560 (IC), and E944 (IIA) and E1005 (IIIA). The overall application of these separate guides and practices is described in Practice E853 (IA)."

ASTM E900 and RG 1.99, Rev. 2 were developed simultaneously under ASTM voluntary consensus due process, The Regulatory Guide DG 1053's methodology requires that:

"The prediction of the vessel fluence must be made by an "absolute" fluence calculation in which the transport of the neutrons from the core is calculated out to the vessel cavity, rather than a simple spatial extrapolation of the fluence measurements."

This is in conflict with the requirements of ASTM E900 and RG 1.99, Rev. 2 and the DG-1053 procedures and methodology have not been subjected to ASTM technical consensus due process.

TABLE 3.1 ATTACHMENT

The following Reference [Mc86] commentary was taken from **Section F, Trend Curve Data Development and Testing**, prepared by W.N. McElroy, R. Gold, E.P. Lippincott, R.L. Simmons and S. Anderson:

"The status of the development and application of new advancements in LWR-PV-SDIP, such as cavity physics-dosimetry for improving the reliability of current and end-of-life (EOL) predictions on the metallurgical conditions of pressure vessels and their support structures, is discussed with appropriate referencing to the current literature, Federal and NRC regulations and rules, and the new series of 21 ASTM LWR Surveillance Standards. Application of established ASTM standards is expected to permit the reporting of measured materials property changes and neutron exposures to an accuracy and precision within bounds of 10% to 30%, depending on the measured metallurgical variable and neutron environment.

NRC Physics-Dosimetry Compendium

The NRC physics-dosimetry compendium (Mc87a) is a collation of information and data developed from available research and commercial light water reactor vessel surveillance program (RVSP) documents and related surveillance capsule reports. The data represents the results of the HEDL (Simons) least-squares FERRET-SAND II Code re-evaluation of exposure units and values for 47 PWR and BWR surveillance capsules for W, B&W, CE, and GE power plants (see Figure HEDL-22). Using a consistent set of auxiliary data and dosimetry-adjusted reactor physics results, the revised fluence values (Table HEDL-22) for E > 1 MeV averaged 25% higher than the originally reported values. The range of fluence values (new/old) was from a low of 0.80 to a high of 2.38.

These HEDL-derived FERRET-SAND II exposure parameter values are being used for NRC supported HEDL and other PWR and BWR trend curve data development and testing studies. These studies are providing results to support Revision 2 of Regulatory Guide 1.99. The information in the compendium is also being made available to the ASTM E10 Committee, to the Metal Properties Council (MPG) Subcommittee 6 on Materials for Nuclear Reactors, and to others developing improved data bases and trend curves. These curves are used by the utilities and by the NRC to account for neutron radiation damage in setting pressure/temperature limits, in making fracture analysis, and in predicting neutron induced changes in reactor PV steel fracture toughness and embrittlement during the vessel's service life.

As stated in Subsection "Key Issues" under "Fluence Issue":

"A survey of the fluence results for the A302B correlation monitor materials made it apparent that the Guthrie data, which employed the NUREG 3319 re-evaluated fluences, provided a better trend curve, indicating the need for, and the potential benefit to be gained from, a self consistent basis for expressing capsule fluence. The impact of the reevaluation on individual data from the Odette et al analysis are apparent.

Many industry neutronics analysts have argued that such dramatic effects on fluence evaluation are unlikely to exist in more recent surveillance data, which "should" have been

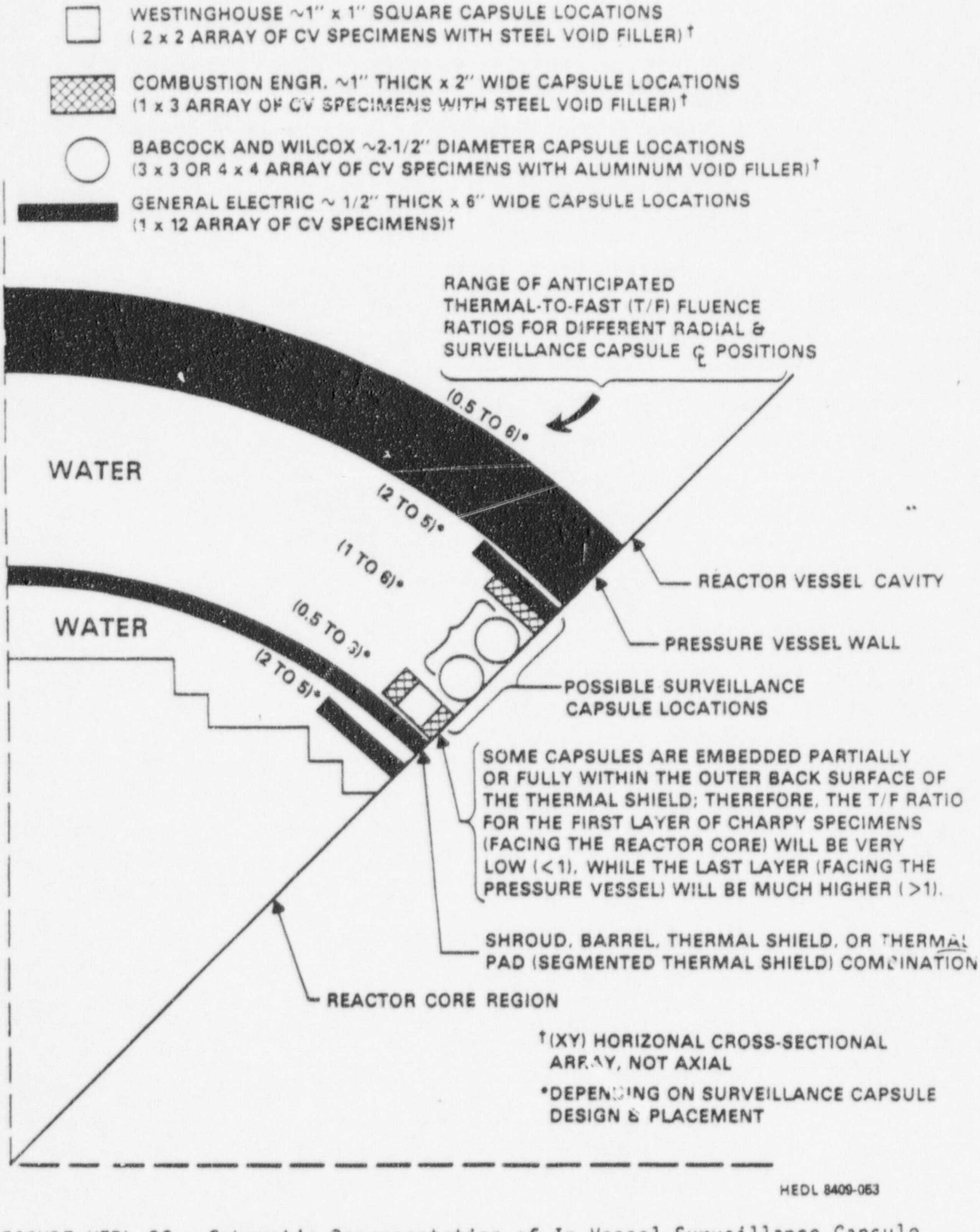

FIGURE HEDL-22. Schematic Representation of In-Vessel Surveillance Capsule Designs and Locations for Operating PWRs and BWRs.

TABLE HEDL-22

RE-EVALUATED EXPOSURE VALUES AND THEIR UNCERTAINTIES FOR LWR-PV SURVEILLANCE CAPSULES

Plant	Unit	Cap-sule	Service Lab*	Biblio Ref	Fluence (ϕt > 1 MeV) (n/cm^2) Old	New [% (1σ)]	New/Old	Fluence (E < 0.414 eV) (n/cm^2)	dpa [% (1σ)]	New dpa/ϕt	dpa/s	hpa (appb)†	Exposure** Time (s)
Westinghouse						***							
Conn. Yankee		A	BMI	(Ir70)	2.08 E+18	3.16 E+18 (12)	1.53	2.54 E+18 (18)	0.00482 (12)	1.52 E-21	9.06 E-11	6	5.233 E+07
Conn. Yankee		F	BMI	(Pe72)	4.04 E+18	6.06 E+18 (24)	1.50	5.43 E+18 (32)	0.00949 (27)	1.56 E-21	1.24 E-10	13	7.651 E+07
Conn. Yankee		H	W	(Ya67)	1.79 E+19	2.00 E+19 (24)	1.12	2.33 E+19 (19)	0.0324 (27)	1.62 E-21	1.36 E-10	52	2.390 E+08
San Onofre		A	SwRI	(No71)	1.20 E+19	2.86 E+19 (22)	2.38	2.05 E+19 (23)	0.0486 (27)	1.70 E-21	8.35 E-10	43	5.824 E+07
San Onofre		D	SwRI	(No72)	2.36 E+19	5.62 E+19 (26)	2.38	3.76 E+19 (23)	0.0944 (29)	1.68 E-21	1.06 E-09	80	8.881 E+07
San Onofre		F	W	(Ya79)	5.14 E+19	5.73 E+19 (14)	1.11	2.99 E+19 (28)	0.0955 (20)	1.67 E-21	3.92 E-10	73	2.438 E+08
Turkey Point	3	S	SwRI	(No79)	1.41 E+19	1.62 E+19 (24)	1.15	1.34 E+19 (24)	0.0255 (27)	1.57 E-21	2.33 E-10	33	1.095 E+08
Turkey Point	3	T	W	(Ya75)	5.68 E+18	7.01 E+18 (10)	1.23	5.12 E+18 (58)	0.0109 (12)	1.55 E-21	4.73 E-10	14	2.302 E+07
Turkey Point	4	S	SwRI	(No79)	1.25 E+19	1.31 E+19 (25)	1.05	1.31 E+19 (25)	0.0213 (27)	1.63 E-21	1.97 E-10	37	1.079 E+08
Turkey Point	4	T	SwRI	(No76)	6.05 E+18	7.54 E+18 (13)	1.25	8.40 E+18 (21)	0.0130 (13)	1.72 E-21	3.48 E-10	20	3.728 E+07
H. B. Robinson	2	S	W	(Ya73)	3.02 E+18	3.91 E+18 (24)	1.29	8.81 E+18 (18)	0.00615 (27)	1.57 E-21	1.06 E-10	19	4.209 E+07
H. B. Robinson	2	V	SwRI	(No76b)	4.51 E+18	7.24 E+18 (22)	1.61	8.96 E+18 (20)	0.0119 (25)	1.59 E-21	1.09 E-10	21	1.050 E+08
Surry	1	T	BMI	(Pe75)	2.50 E+18	2.86 E+18 (9)	1.14	3.57 E+18 (20)	0.00449 (12)	1.57 E-21	1.33 E-10	8	3.378 E+07
Surry	2	X	BMI	(Pe75a)	3.02 E+18	3.03 E+18 (11)	1.00	3.64 E+18 (20)	0.00473 (13)	1.56 E-21	1.28 E-10	9	3.687 E+07
North Anna	1	V	B&W	(Lo81d)	2.49 E+18	2.72 E+18 (9)	1.09	5.80 E+18 (14)	0.00411 (11)	1.51 E-21	1.15 E-10	11	3.570 E+07
Beznau	2	R	EIR	( )	1.70 E+19	1.34 E+19 (9)	1.27	2.27 E+19 (21)	0.0198 (11)	1.48 E-21	1.16 E-10	49	1.714 E+08
Pr. Island	1	V	W	(Da77)	5.21 E+18	6.03 E+18 (11)	1.16	9.21 E+18 (21)	0.0102 (16)	1.69 E-21	2.41 E-10	20	4.248 E+07
Pr. Island	2	V	W	(Ya81)	5.49 E+18	6.74 E+18 (10)	1.23	9.75 E+18 (26)	0.0117 (13)	1.74 E-21	2.67 E-10	21	4.394 E+07
R. E. Ginna	1	R	W	(Ya74)	7.60 E+18	1.17 E+19 (10)	1.54	1.84 E+19 (25)	0.0215 (14)	1.83 E-21	2.59 E-10	38	8.328 E+07
R. E. Ginna	1	V	W	(Ma73a)	4.90 E+18	5.93 E+18 (14)	1.21	1.37 E+19 (59)	0.0102 (22)	1.72 E-21	2.20 E-10	29	4.612 E+07
Kewaunee		V	W	(Ya77)	5.59 E+18	6.41 E+18 (10)	1.15	1.23 E+19 (23)	0.0114 (13)	1.78 E-21	2.82 E-10	26	4.057 E+07
Point Beach	1	S	W	(Ya76)	7.05 E+18	8.45 E+18 (10)	1.20	1.20 E+19 (19)	0.0146 (13)	1.73 E-21	1.25 E-10	27	1.163 E+08
Point Beach	1	R	W	(Ya78)	2.22 E+19	2.29 E+19 (10)	1.37	2.85 E+19 (22)	0.0408 (13)	1.78 E-21	2.50 E-10	61	1.632 E+08
Point Beach	2	V	BMI	(Pe75b)	4.74 E+18	7.28 E+18 (11)	1.54	1.09 E+19 (18)	0.0121 (13)	1.66 E-21	2.52 E-10	23	4.805 E+07
Point Beach	2	T	W	(Da78a)	9.45 E+18	9.40 E+18 (10)	0.99	1.48 E+19 (21)	0.0157 (12)	1.67 E-21	1.44 E-10	32	1.087 E+08
Point Beach	2	R	W	(Ya79a)	2.01 E+19	2.52 E+19 (10)	1.25	4.71 E+19 (26)	0.0460 (14)	1.83 E-21	2.81 E-10	93	1.640 E+08
D. C. Cook	1	T	SwRI	(No77b)	1.80 E+18	2.71 E+18 (22)	1.51	3.26 E+19 (19)	0.00445 (25)	1.64 E-21	1.12 E-10	77	3.991 E+07
Indian Point	2	T	SwRI	(No77a)	2.02 E+18	3.28 E+18 (22)	1.62	4.01 E+18 (44)	0.00537 (27)	1.64 E-21	1.20 E-10	91	4.473 E+07
Indian Point	3	T	W	(Da79)	2.92 E+18	3.23 E+18 (22)	1.11	3.13 E+18 (21)	0.00520 (25)	1.61 E-21	1.23 E-10	74	4.211 E+07
Zion	1	T	BMI	(Pe78)	1.80 E+18	3.04 E+18 (10)	1.69	3.17 E+18 (21)	0.00488 (12)	1.61 E-21	1.29 E-10	82	3.789 E+07
Zion	1	U	W	(Ya81a)	8.92 E+18	1.01 E+19 (10)	1.13	8.87 E+18 (24)	0.0166 (13)	1.64 E-21	1.47 E-10	21	1.123 E+08
Zion	2	U	BMI	(Pe78)	2.00 E+18	2.80 E+18 (9)	1.40	3.80 E+18 (15)	0.00446 (12)	1.59 E-21	1.11 E-10	10	4.007 E+07
Salem	1	T	W	(Ya80)	2.56 E+18	2.84 E+18 (22)	1.11	3.26 E+18 (19)	0.00460 (25)	1.62 E-21	1.34 E-10	7	3.426 E+07

*BMI = Battelle Memorial Institute; W = Westinghouse; SwRI = Southwest Research Institute; CE - Combustion Engineering; ET = Effects Technology; B&W = Babcock and Wilcox; EIR = Eidg. Institute für Reaktorforschung.

**Equivalent constant power level exposure time.

***3.16 E+18 (12) means 3.16 x 10^{18} with a 12% (1σ) uncertainty.

†Calculated for A302B steel with a nominal concentration of 0.55 appm boron present.

TABLE HEDL-22 (Cont'd)

Plant	Unit	Capsule	Service Lab*	Biblio Ref	Fluence (φt > 1 MeV) (n/cm^2) Old	Fluence (φt > 1 MeV) (n/cm^2) New [% (1σ)]	Fluence (φt > 1 MeV) (n/cm^2) New/Old	Fluence (E < 0.414 eV) (n/cm^2)	dpa [% (1σ)]	New dpa/φt	dpa/s	hpa (appb)†	Exposure** Time (s)
Combustion Engineering													
Palisades		A240	BMI	(Pe79b)	4.40 E+19	6.06 E+19 (23)	1.38	7.26 E+19 (61)	0.0972 (28)	1.60 E-21	1.36 E-09	170	7.130 E+07
Fort Calhoun		W225	CE	(By80)	5.10 E+18	5.83 E+18 (14)	1.14	3.09 E+19 (60)	0.00879 (18)	1.51 E-21	1.07 E-10	63	8.191 E+07
Maine Yankee		1	ET	(Wu75)	1.30 E+19	1.76 E+19 (19)	1.35	3.00 E+19 (29)	0.0285 (23)	1.62 E-21	1.03 E-09	62	2.777 E+07
Maine Yankee		2	W	(Ya81b)	8.84 E+19	7.73 E+19 (13)	0.87	1.20 E+20 (23)	0.121 (18)	1.57 E-21	8.38 E-10	230	1.446 E+08
Maine Yankee		W263	BMI	(Pe80)	7.10 E+18	5.67 E+18 (12)	0.82	2.67 E+19 (21)	0.00843 (14)	1.49 E-21	5.83 E-11	55	1.446 E+08
Babcock & Wilcox													
Oconee	1	F	B&W	(Lo75)	8.70 E+17	6.98 E+17 (21)	0.80	1.00 E+18 (13)	0.000959 (19)	1.37 E-21	3.65 E-11	3	2.629 E+07
Oconee	1	E	B&W	(Lo77)	1.50 E+18	1.50 E+18 (10)	1.00	2.61 E+18 (15)	0.00208 (10)	1.39 E-21	4.01 E-11	7	5.186 E+07
Oconee	2	C	B&W	(Lo77a)	9.43 E+17	1.01 E+18 (10)	1.07	1.55 E+18 (15)	0.00148 (11)	1.47 E-21	3.88 E-11	4	3.802 E+07
Oconee	3	A	B&W	(Lo77b)	7.39 E+17	8.05 E+17 (10)	1.09	1.34 E+18 (11)	0.00113 (11)	1.40 E-21	3.79 E-11	3	2.983 E+07
Three Mile Is.	1	E	B&W	(Lo77c)	1.07 E+18	1.09 E+18 (9)	1.02	1.90 E+18 (11)	0.00151 (9)	1.39 E-21	3.75 E-11	5	4.036 E+07
Arkansas Nuclear	1	E	B&W	(Lo77d)	7.27 E+17	8.18 E+17 (8)	1.13	6.32 E+17 (9)	0.00117 (8)	1.43 E-21	3.92 E-11	2	2.981 E+07
General Electric													
Dresden	3	4G14	W	(Ya82)	2.06 E+19	1.86 E+19 (17)		1.51 E+20 (62)	0.0285 (17)	1.53 E-21	3.35 E-10	290	8.483 E+07
Dresden	3	4G15	W	(Ya82)	1.50 E+19	1.35 E+19 (17)		1.19 E+20 (62)	0.0209 (17)	1.55 E-21	2.46 E-10	240	8.483 E+07
Dresden	3	4G16	W	(Ya82)	1.20 E+19	1.08 E+19 (17)	0.89	9.70 E+19 (62)	0.0168 (17)	1.55 E-21	1.98 E-10	200	8.483 E-07
Dresden	3	4G17	W	(Ya82)	5.16 E+18	4.51 E+18 (17)		5.20 E+19 (62)	0.00733 (18)	1.63 E-21	8.64 E-11	120	8.438 E+07
Quad Cities	1	3G5	W	(Ya81c)	4.04 E+19	4.23 E+19 (17)		2.41 E+20 (62)	0.0604 (17)	1.43 E-21	4.85 E-10	400	1.243 E+08
Quad Cities	1	3G6	W	(Ya81c)	3.08 E+19	3.12 E+19 (17)		1.92 E+20 (62)	0.0450 (17)	1.44 E-21	3.62 E-10	340	1.243 E+08
Quad Cities	1	3G7	W	(Ya81c)	2.37 E+19	2.47 E+19 (17)	1.01	1.54 E+20 (62)	0.0356 (17)	1.44 E-21	2.86 E-10	290	1.253 E+08
Quad Cities	1	3G8	W	(Ya81c)	1.24 E+19	1.17 E+19 (17)		1.03 E+20 (62)	0.0180 (17)	1.54 E-21	1.45 E-10	210	1.243 E+08
Quad Cities	2	3G14	W	(Ya82a)	4.14 E+19	4.28 E+19 (16)		2.45 E+20 (62)	0.0611 (17)	1.43 E-21	4.29 E-10	400	1.422 E+08
Quad Cities	2	3G15	W	(Ya82a)	3.48 E+19	3.60 E+19 (16)		2.13 E+20 (62)	0.0516 (17)	1.43 E-21	3.63 E-10	370	1.422 E+08
Quad Cities	2	3G16	W	(Ya82a)	2.43 E+19	2.52 E+19 (16)	1.03	1.55 E+20 (62)	0.0362 (17)	1.44 E-21	2.54 E-10	290	1.422 E+08
Quad Cities	2	3G17	W	(Ya82a)	2.32 E+19	2.37 E+19 (17)		1.49 E+20 (62)	0.0342 (17)	1.44 E-21	2.41 E-10	290	1.422 E+08
							Avg 1.25						

*BMI = Battelle Memorial Institute; W = Westinghouse; SwRI = Southwest Research Institute; CE = Combustion Engineering; ET = Effects Technology; B&W = Babcock and Wilcox; EIR = Eidg. Institute für Reaktorforschung.
**Equivalent constant power level exposure time.
***3.16 E+18 (12) means 3.16×10^{18} with a 12% (1σ) uncertainty.
Calculated for A302B steel with a nominal concentration of 0.55 appm boron present.

evaluated on a self-consistent basis. Interestingly, including more recent surveillance data from the PREDB for the correlation monitor material with data in Rev. 2 increases the scatter alarmingly, suggesting that the more modern data are no more consistent than the original RG 1.99, Rev. 2 surveillance data. It is therefore concluded that an uncritical statistical analysis of the raw surveillance data base is unlikely to reduce data scatter and hence the margins. On the contrary, it is more likely to increase the margins. This would seem to be a strong argument for continuation of the exercise initiated in NUREG 3319."

Fortunately for the U.S. Nuclear Industry, Lippincott and Anderson have continued the NUREG 3319 exercise for Westinghouse surveillance capsules and have carefully reported and documented the results [Li94,Li96]. They state [Li96]:

"A systematic analysis of dosimetry from Surveillance capsules from Westinghouse plants has shown that excellent consistency of results is obtained. The indicated precision of results for both calculations and measurements falls well within the limits required for regulatory compliance.

.......the measured surveillance capsule results for all the domestic Westinghouse plants were reanalyzed using a consistent methodology to derive the fast neutron fluence (E > 1 MeV). The study included a total of 131 capsules which were originally analyzed between 1970 and early 1994. The comparison of calculated fluence for these capsules with the measured results provides a large body of results for establishment of the precision of fluence determination. ....In the reanalysis of capsule fluences from dosimetry measurements, an approach slightly different from that typically used for the analysis of individual dosimetry sets was adopted in order to maximize the consistency of the results. The assumption was made that variation of the neutron spectrum for capsules in symmetric locations in the same or similar plants is small compared to uncertainties in the measured reaction rates. The analysis procedure was then carried out using the following steps;

1) Reevaluate the dosimetry reaction rates for all capsules.

2) Average normalized reaction rates for similar capsules.

3) Derive a best estimate spectrum for each capsule location using the average reaction rates.

4) Calculate a measured fluence (E > 1 MeV) for each capsule using a weighted average of the fluence indicated by each threshold dosimeter."

Lippincott states in Ref. [Li94]:

"Almost all of the capsules with large changes from the original values were included in those analyzed by Simons and reported in NUREG 3319. Comparisons with Simon's results are presented in Table .... This table contains the original values for fluence (E > 1 MeV), the updated values from the work reported in this document, and Simon's values. Most of the updated fluence values are in reasonable agreement with Simons. The exception is the values for San Onofre Unit 1 where Simons is substantially higher. The difference probably arises form the spectrum used to derive the fluence from the measurements and to the lack

of fission monitor data. The comparisons with the calculated fluences for these capsules indicates that the values in this report are probably correct. On the average, the fluence results from Simons (~ 1985) are 32% higher than the original values, and 11% higher than the Westinghouse *(~ 1994)* updated values. Expected bias between the Simons fluences and the updated values is about 5% due to the change in fluence location from the center of the capsule to the center of the charpy specimens."

Lippincott and Anderson have summarized the results of their updated studies in Table A-7, taken here from Ref. [Li94]. In Ref. [Li96], they state:

"For all the capsules taken together, the C/M ratio averages 0.88 with a standard deviation of 10%. However, the consistency of results for individual plant types is below this 10% value in all cases. For the newer plant designs (the plants with neutron pads), which are expected to have the most accurate results, the standard deviations is about 5%. This value of 5% can be assumed to represent a reasonable estimate of the precision of fluence analysis for capsules with well designed dosimetry sets.

It is also seen from the results in Table A-7 that some of the capsules from thermal shield plants fall outside the expected bounds based on this precision estimate. Two early capsules from the two-loop plants had less dosimetry coverage and were analyzed by a different laboratory from the rest. Exclusion of these two capsules results in the same consistency of results from the remaining 19 capsules as obtained from the later plants. It is also noted that two four-loop plants have consistent results with larger bias, possibly due to a structural difference in surveillance capsule location. With these exceptions, the three-loop thermal shield plants exhibit the greatest variation, but the results for individual plants are generally consistent with the 5% precision value observed in the newest plants.

The reanalysis of measured data for a group of capsules using the latest nuclear data libraries based on ENDF/B-VI resulted in no significant change to the measured fluence values. This is expected since changes to the dosimetry cross-sections were small and any changes in the calculated neutron spectrum are minimized by the FERRET adjustment procedure. The calculated fluence values using the BUGLE-93 transport cross-sections did increase significantly, however, compared to the SAILOR results. The change in calculated fluence values resulted in an increase in the average C?M bias from 0.88 to 1.07. Thus the new cross sections result in a smaller bias, but seem to overcorrect the previously observed bias. Using new cross sections, the calculated fluence values may be expected to be conservative for predictions of radiation effects. The change in bias occurred consistently for the capsules studied, and it is concluded that the standard deviation of the C/M result given in Table A-7 would not be significantly different if all the capsules were evaluated using the newer (ENDF/B-VI) cross sections."

In conclusion, Lippincott and Anderson [Li96] state:

"The consistency of the C/M results over the large number of fuel arrangements and types of plants indicates that the calculation of both capsule and vessel fluences can be made with high precision, and after bias is taken into account, high accuracy can be obtained also.

Table A-7

Summary of C/M Results

Plant Type	Number of Capsules	C/M	Standard Deviation (%)
Two Loop Plants	21	0.767	8.9
Three TS Loop Plants:			
Beaver V. 1, N. Anna	7	1.0[illegible]7	5.8
Surry	6	0.903	7.6
Turkey Point	5	0.823	2.7
Robinson	3	0.937	4.6
All Three Loop TS Plants	21	0.927	9.9
Three Loop NP Plants	12	0.916	5.1
Four Loop TS Plants:			
Sequoyah 1	3	0.748	5.4
Zion 1	4	0.780	8.8
Other 4 Loop TS Plants	29	0.924	5.4
All Four Loop TS Plants	36	0.893	9.1
Four Loop NP Plants	33	0.868	4.9
Connecticut Yankee	5	0.989	8.8
San Onofre 1	3	0.967	7.4
All Plants	131	0.879	10.3

Notes: C/M is ratio of calculated fluence (E > 1 MeV) to that derived from the dosimetry measurements at the center of the specimens in each capsule. TS indicates thermal shield plants and NP indicates neutron pad plants. Connecticut Yankee and San Onofre 1 have different geometry and are not included with the other thermal shield plants.

In particular, the consistency of results indicates that the calculations are adequately taking into account the variation due to the change in fuel loading, including high burnup and low leakage fuel management. The calculation of fluence for fuel cycles beyond the time of the latest capsule removal can therefore be relied upon to produce good estimates of relative exposure. The consistency also indicates the high quality of the measurement results. It is concluded that even a single capsule measurement result can be used to significantly reduce fluence uncertainty by defining plant specific bias.

Use of the BUGLE-93 cross section set results in a significant change in the average capsule bias but little change in data scatter. Therefore, all the conclusions regarding the precision of experimental and calculational results drawn on the basis of the older cross section calculations are still valid. In addition, the values of the measured fluence for surveillance capsules derived using the older cross section set are still valid."

Contiuning the commentary taken from **Section F, Trend Curve Data Development & Testing:**

Regulatory Guide 1.99, Revision 2

"In Ref (Ra84), Randall discusses the basis for Revision 2 of Reg. Guide 1.99. As stated, the Guide is being updated to reflect recent studies of the physical basis for neutron radiation damage and efforts to correlate damage to chemical composition and fluence. Revision 2 contains several significant changes. Welds and base metal are treated separately. Nickel content is added as a variable, and phosphorus is removed. The exponent in the fluence factor is reduced, especially at high fluences; and guidance is given for calculating attenuation of damage through the vessel wall.

For PV wall neutron fluence attenuation predictions, the preliminary results of the PSF (Mc85) comparisons lie within 10% but reaffirm slight deficiencies in the iron cross sections first brought to light by the PCA and PSF startup experiment comparisons (Mc81, Wi83), which show increasing disagreement the further into the PV one goes.

In the planned Revision 2 of Reg. Guide 1.99 (Ra84), the equation used for PV wall fluence attenuation by Randall is

$$Fluence(x) = Fluence(Surface) \cdot e^{-0.24x} \qquad (1)$$

where x is the depth in the wall in inches, measured from the inside surface. This equation is based on transport calculations by Guthrie et al. (Gu82, Gu82a) for the dpa attenuation through an 8.0-inch vessel wall. These calculations did not account for the deficiencies in the iron cross sections mentioned above.

It has been recently noted by Fabry that the $Li(n,\alpha)$ spectrometry data (DeLeeuw, in Mc81) in PCA are consistent with gas proton recoil spectrometry (Rogers, in Mc81) and silicon damage measurements (DeLeeuw, in Mc81), and they indicate larger proportions of neutrons below 1.0 MeV than predicted by ENDF/B-IV; the discrepancy is on the order of 20%, in the same direction as nuclear research emulsion (NRE) results reported by Roberts, Gold, and Preston in Ref (Mc85), Section 2.2.1.1, NRE Measurements. This confirmed result does affect the $dpa/\Phi > 1$ MeV transverse predictions through the reactor PV planned for use in

Reg. Guide 1.99, Revision 2 (Ra84), and may adversely impinge upon eventual crack-arrest considerations in the safety analysis of ASME-III designed vessels. It is recommended, therefore, that:

1) A new simultaneous evaluation of all experimental data in PCA, the NESDIP replica, and the Mol Iron Shell Benchmarks should be performed, including the French damage monitor results obtained during the PSF startup program,

2) Integral measurements using NRE as well as higher threshold-energy sensors [such as $^{58}Ni(n,p)$, $^{64}Zn(n,p)$, or $^{27}Al(n,\alpha)$] should be performed in the Mol Iron Shell Benchmarks, and

3) Continuous gamma-ray spectrometry experiments should be conducted in the NESDIP benchmark, Phase 3, to resolve inelastic gamma-rays produced by fast neutron interactions in iron and there by test the inelastic neutron transport cross section of iron."

The above confirmed result has been re-confirmed by the LR-O Benchmark studies of Osmera et al.; see the Section 9, Subsection 9th ASTM-EWGRD Symposium, Paper 3 commentary. Using the ENDF/B-VI data base, BUGLE-93 library with the 1D ANISN and 2D Dort transport codes, Osmera et al. state:

"The spectra measured by both spectrometers were identical in the frame of usual uncertainties known from the measurement in reference fields. The proton recoil spectrometers' experimental (higher upper energy limit) and 2D-DORT, BUGLE-93 Skoda calculation were used for the comparison. The sensitivity of the results to the substitution of SAILOR BY BUGLE-93, P7, was studied with 1D (ANISN) calculation. ***The differences (improvements) were remarkable. Nevertheless the discrepancy of the 2D-DORT, BUGLE-93, calculation and experiment is substantial for the evaluation of the reliability of the RPV exposure calculation............. The calculation mostly underestimated the fast fluxes.***

...

.... ***The disagreement of calculation and experiment is probably caused by the used group library. In several studies the sensitivity of the results to the type of the library are presented [Ha96a,Ch92]. The Monte Carlo calculations of similar problems [Osxx,Woxx] show substantially better consistency with experiment."***

Additional support for these conclusions is given by Hogel et al. in their Prague papers [Ho96,Ho96a] on "Neutron Dosimetry in Extended Surveillance (Capsule) Program on the 4th Unit of NPP Dukovany" and "Fast Neutron Fluence Monitoring (Cavity) on NPP Dukovany;" see Subsection 9, 9th ASTM-EWGRD Symposium, Papers 4 and 5, respectively.

What this shows is that the results of the application of the procedures, methodology, and data files recommended in the ASTM E1018 Dosimetry Cross Section File standard are of such high quality, that the use of the current ENDF/B-VI, or an earlier version, of the files with one of the codes listed in the ASTM E944 Adjustment Code Standard will provide extremely reliable "best estimate" values of fluence for NPP surveillance and regulatory programs.

KEY ISSUES -- Detailed commentary is provided on important technical issues with relevance to the PNP's Physics-Dosimetry Program and the use of the recommended methodology in the ASTM E706 LWR Surveillance Standards; see **Attachment II** (of the March 30, 1996 ASTM E10.05.04 TG Report) on "Technical Issues Relevant to LWR Surveillance Standards" and its "Reference Listing" and **Attachment III** on "ASTM Standards Associated with PWR and BWR Power Plant Licensing, Operation and Surveillance" and its "Reference Listing." Issue statements from Attachments II and III with relevance to the present review and study are:

Benchmark Field (BF) Issue [3]

The issue of the limitations of BF is addressed by Gold in Refs. [7,8]. Lippincott [14] found that because of the large potential impact of the uncertainties in plant specific parameters (reactor geometry etc.) use of only one or two reactors to benchmark plant calculations and determine generic biases in the calculational methodology and cross sections is not adequate.

In Ref. [13], Blaise, de Wouters, and Ait Abderrahim found that further away from the (VENUS) core, the average C/E in the neutron pad from 12 measurements is 1.01 ± 0.11 (s.d). In this study, MCBEND provided an essentially unbiased approximation of the "equivalent fission flux" from the core baffle to the neutron pad, without significant variation of C/E in the range of penetration depth. This conclusion generally confirms previous validation work in slab geometry benchmark fields with a pure U-235 fission source spectrum. This encouraging result, obtained with a realistic core shape and a spectrum of emerging neutrons similar to that in a NPP, has confirmed the choice of MCBEND as a suitable tool for the computation of the fast fluence at the PV and the surveillance capsules of seven Belgian PWRs presently in operation.

Transport Calculational Issue

A number of past and present Symposia papers address the issue of the use of the Discrete Ordinates and Monte Carlo Methods. Of interest here are the conclusions of Gold's study of the "Limitations of Pressure Vessel Fluence Calculations" [7,8], Helm's review of "Past and Current Methods Used for Fluence Spectrum Estimation" [15,16], Lippincott's review of the "Assessment of Uncertainties in RPV Fluence Determination" [14], and de Wouters et al.'s study of the "Analysis of PWR PV Surveillance Dosimetry with MCBEND" [17].

Gold concludes that given the current limitations that exist in calculations, the only rational way to determine PV and SS neutron fluence with the necessary accuracy and reliability for evaluating and predicting steel radiation damage is through reliance upon experiment. This can be accomplished by application of a least squares adjustment code, as described in ASTM Standard E944 [E706(IIA)], which judiciously combines calculational and experimental results. Gold uses the existing calculational limitations to show that the recent draft RG DG 1053 for the determination of pressure vessel neutron fluence is based upon procedures and

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[3] The numbered references can be found in Attachments II and III.

assumptions that are **not** valid. Gold also emphasizes that the limitations possessed by experimental methods must be carefully considered and evaluated. As a consequence, specific and detailed recommendations to improve LWR-PV-SS surveillance dosimetry have been advanced elsewhere [8].

In regard to the use of the Monte Carlo method, de Wouters et al. [17] found that the MCBEND code provides an accurate approximation of fast fluence at the level of surveillance capsules and suggest that it is a suitable method to calculate the fast fluence at the inner surface of the PWR PV with little or no bias.

Exposure and Radiation Damage Parameter Extrapolation Issue

The complexities and limitations of PV neutron fluence calculations make alternative methods of neutron fluence extrapolation desirable. To overcome uncertainties and systematic biases that can be introduced by calculational methods, an empirical two step method of extrapolation has been advanced by Gold and McElroy [22,23].

Least-squares analyses of the Pool Critical Assembly (**PCA**) and Pool Side Facility (**PSF**) benchmark dosimetry data have demonstrated that a simple exponential description of either the DPA or fluence is an excellent representation of the radial variation of neutron exposure within the PV wall. This simple behavior can be used to advantage in obtaining extrapolated exposure and damage parameters (such as the Charpy shifts) at points of interest within the PV. The ASTM Standard on Extrapolating PV Surveillance Dosimetry Results, E706(IC), should be revised to indicate the availability of this new empirical extrapolation alternative.

Regulatory Guide 1.99, Revision 2 Data Base Consistency Issues

The methodology and data used in RG 1.99, Rev. 2 for predicting Charpy shifts are, essentially, the same as those used and recommended in the ASTM E900-87 [E706(IIF.1)] Standard. E900-87, however, does not address the issue of ΔRT_{NDT} through wall attenuation.

The following critique related to RG 1.99 Rev. 2 consistency issues is based, primarily, on the results of recent studies by R. McElroy. [4] An important part of this work is associated with an in-depth review of the methodology to be considered and recommended for use in future versions of E706(IE), E706(IIF.1), E706(IIF.2), E706(IIF.3) and E706(IIID.2).

The two separate data bases employed to produce RG 1.99, Rev. 2 were quite different. Two independent trend curves, for both plate and weld, one by Odette et al. and the other by Guthrie, were combined to provide the most conservative elements of each in a final form embodied in Rev. 2 [51,52].

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[4] Symposium Oral Session A presentation by R.J. McElroy on "Embrittlement Trend Curve Development Issues."

Fluence Issue -- For the two data bases, the fluences differ in the majority of cases, some by more than a factor of two. The reason for this difference is that Odette et al, most often, employed the fluences quoted in surveillance reports and compiled in the EPRI data base [53], while Guthrie employed a re-evaluated set of fluences from NUREG-3319 produced by Simons, McElroy, and Lippincott under the LWR-PV-SDIP [36] as well as some values taken directly from surveillance capsule reports. The 3319 fluences were on the average 25% higher than the original fluence estimates employed by Odette et al., some being as much as 2.5 times higher.

A survey of the fluence results in the PREDB for the A302B correlation monitor materials made it apparent that the Guthrie data, which employed the NUREG 3319 re-evaluated fluences, provided a better trend curve, indicating the need for, and the potential benefit to be gained from, a self consistent basis for expressing capsule fluence. The impact of the reevaluation on individual data from the Odette et al analysis are apparent.

Many industry neutronics analysts have argued that such dramatic effects on fluence evaluation are unlikely to exist in more recent surveillance data, which "should" have been evaluated on a self-consistent basis. Interestingly, including more recent surveillance data from the PREDB for the correlation monitor material with data in Rev. 2 increases the scatter alarmingly, suggesting that the more modern data are no more consistent than the original Rev. 2 surveillance data. It is therefore concluded that an uncritical statistical analysis of the raw surveillance data base is unlikely to reduce data scatter and hence the margins. On the contrary, it is more likely to increase the margins. This would seem to be a strong argument for continuation of the exercise initiated in NUREG 3319.

Charpy Shift Issue -- The next inconsistency between the two data bases arises in the Charpy shifts. In this case, Guthrie used the shifts quoted in the surveillance reports, which were largely judged "by eye", or where no 30 ft-lb result was reported, Randall and Guthrie made their own estimate of the shift from the plotted data. Odette et al, on the other hand, used shifts based on tanh fitted curves to the EPRI data base. Virtually all Charpy shifts differ between the two data bases and in approximately 15% of the cases this difference is substantial.

Copper and Nickel Issue -- Differences in Cu and Ni content are also apparent between the two data bases, though these are less common and therefore of less importance. It is now clear that bulk Cu content is a poor indicator of the potential for embrittlement. It can be demonstrated with the PREDB that, for welds with Cu content above about 0.23wt%, the property behavior is effectively independent of Cu content, as might be expected, since the Cu solubility at a typical RPV Post-Weld Heat Treatment (**PWHT**) temperature of 607 ± 14°C is about this level.

It should be noted that widely accepted techniques are now available for the measurement of soluble copper as well as for monitoring the copper precipitation during irradiation. Such techniques would almost certainly demonstrate a strong "embrittlement saturation effect", particularly at the higher irradiation temperatures where matrix hardening would be reduced by recovery effects.

The role of Ni could be more complex over the range of Cu and Ni contents encountered in the PREDB and it may be necessary to further subdivide materials into low and high Ni groups as well as into plate and weld, which is the basis of the current Regulatory Guide. Indeed, further subdivision into low and high Cu might also be warranted due to the non-uniform distribution, especially in welds, of Cu and Ni.

A simple form of the trend curve describing the above behavior would contain a Cu hardening component which increases with Cu up to a level judged to be representative of an upper bound solubility based on the vessel's heat treatment. Above this Cu level, the Cu hardening component would be constant and independent of Cu content.

Neutron Flux or Damage Rate Issue -- Another effect which is of potential importance, particularly in relation to BWR's and accelerated Test Reactor (**TR**) irradiations, is the effect of neutron flux or damage rate. The copper hardening component increases during irradiation to a plateau level corresponding to full copper precipitation. The plateau level increases with increasing copper content in solid solution.

There is now considerable evidence that the copper precipitation process, which is irradiation enhanced, is a function of both fluence and time, such that at low dose rates full precipitation is achieved at lower doses. Neutron flux or damage rate should therefore be included in any reassessment of embrittlement trend curves, since surveillance exposure rates in the order of $3 \times 10^{8} n/cm^{2} \cdot s$ are to be found in the PREDB, and if TREDB irradiations are to be included, they can be as high as $10^{13} n/cm^{2} \cdot s$. Stallmann et al will be reporting at this symposium on the status of development of the TR Embrittlement Data Base [34].

Consistency of Methodology Issue -- It is apparent that significant differences existed between the two data base analyses employed in deriving the Rev. 2 trend curve and that neither followed a consistent methodology. A more consistent approach would have been to combine the Odette et al Charpy data with the Guthrie fluences and to ignore data for which copper and nickel values were in doubt or not available. Such an approach appears not to have been attempted, and it is probable that a more self-consistent approach would have reduced the margins and yielded a better trend curve, even given the unrepresentative nature of the copper contribution.

An NRC sponsored analysis of the PREDB is in progress and it is to be hoped that the object lessons derived from the recognized inconsistencies in the Rev. 2 analysis will be corrected.

There remains the need to review and revise all fluence data employed in the current analysis using self-consistent dosimetry and neutronics methodologies. Without such a revaluation there is evidence that margins will increase significantly.

Additionally, advantage should be taken of mechanistic and microchemical technique developments which should allow more representative estimates of copper and matrix hardening effects to be assigned to account for effects of PWHT, capsule temperatures and damage rate.

Incorporation of all these factors in a strongly phenomenological approach will provide greater confidence in the specification of trend curves, particularly for maximum life attainment of existing NPPs and ALWRs, while maintaining more realistic and definable safety margins.

In regard to the above, the Westinghouse [Li94,Li96] results of the application of the ASTM E706 Master Matrix, self-consistent dosimetry and neutronics methodologies, were submitted for inclusion in the PREDB physics-dosimetry-metallurgy data base. This data base was used for the development and testing of the recently balloted revised **ASTM E900-98 Standard on "Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials."**

In the "Technical Basis of ASTM E900-98" Attachment to the E900-98 Standard's ballot, it states under Item (2), Section 4.0, Data Compilation and Description of Data Base, that:

> *"Fluence values were updated by ABB-CE, General Electric, Westinghouse, and Framatome Technologies."*

It is my understanding, that except for Westinghouse, the other three vendors did not followed the Table 3.1 recommended ASTM procedures and methodology in updating their surveillance capsule fluence values for the PREDB. That is the methodology required under Sections 1.2 of the Scope Sections 1.2 of E900-98 and Section 1.3 of E900-87. It would appear, therefore, that the more consistent and updated Westinghouse surveillance capsule fluence values were not used in the development of the ASTM E900-98 predictive formulas. That is, since un-determined fluence value biases could be introduced into the data base by the use of the combined ABB-CE, General Electric, Westinghouse, and Framatome Technologie results if the same procedures and methodologies for the determination of the fluence values had not been used by all four service organizations.

LMFBR-ILRR-FTR PHYSICS-DOSIMETRY CHARACTERIZATION PROGRAM REQUIREMENTS AND RESULTS -- The reason that the adjustment code methodology is so well established and validated is because the success of the LWR-PV-SDIP program was dependent on the remarkable success of the LMFBR development program [Mc77,Li83]. Relevant excerpts from "Neutron Environmental Characterization Requirements for Reactor Fuels and Materials Development and Surveillance Programs" [Mc77] and the FTR Dosimetry Handbook [Li83] are attached to this letter report. In Mc77, Figure 2 represents an estimate of pre- and post-1975 state-of-the-art nuclear parameter uncertainties for neutron environmental characterization for U.S. reactor development programs. After 1970, the results and projected estimates are primarily based on EBR II and the Interlaboratory LMFBR Reaction Rate (ILRR) program dosimetry test results and reactor physics studies. By 1985, estimated uncertainties were in the range of 5 to 15% (1σ) for such dosimetry measurement adjusted parameters as neutron flux and fluence for E>1 Mev. The basis for the Figure 2 and Table I estimate of 5 to 15% for dosimetry measurement adjusted values of fast flux and fluence (E>1 Mev) was the success of the ILRR program [Mc75]. In [Mc75, page 180], McElroy and Kellogg state:

"The development, design, & operation of nuclear reactors require the accurate prediction of

a) fission rates and burnup for fuels and

b) neutr.. exposure for neutron induced property changes for fuels and materials.

Goal accuracies of as low as 1% (1σ) have been set for the determination of fission rates, burnup, and neutron fluences for the fast reactor development program. Based on the discussion of the status of fuels and materials fast reactor dosimetry data development and testing, attainable goal accuracies presently appear to be in the range of 2 to 5%.

Comparisons are made of CFRMF and ΣΣ results with those reported previously for GODIVA and the ^{235}U fission spectrum. Fission yield results are considered, based on measurements in the high-intensity environment of EBR II. These results are coupled with those from CFRMF, ΣΣ, and other neutron fields to more clearly define and document existing uncertainties associated with reaction-rate, fuel burnup, and flux-spectral-fluence determination for fast reactors.

Information on the application of the SAND II multiple foil method of neutron-flux-spectral characterization, developed for the U.S. Atomic Energy Commission's Fast Reactor Materials Dosimetry Center (FRMDC) at HEDL is presented in Ref. 3. The SAND II Monte Carlo error analysis code [Os76] is used in this paper to assign uncertainties to multiple-foil-derived flux spectra for two reference standard neutron fields, CFRMF and ΣΣ. The foil results are compared with spectrometry and calculations to help identify sources of uncertainties in current estimates of flux spectra for these two important neutron fields.

The value of absolute total flux at 6 kW derived by the SAND II Monte Carlo code for CFRMF is 7.36×10^{10} n/(cm^2 sec) ± 2.3% (1σ).

Using the 36 group calculated spectrum as input, the SAND II Monte Carlo derived value of absolute flux at a reactor power of 1 MW for ΣΣ is 7.4×10^8 n/(cm^2 sec) ± 2.8% (1σ). ... Here too, the multiple-foil-derived spectrum is harder than that calculated."

It is important to understand that in 1975 (and it is still true today), the results of the ILRR [Mc75], EBR II & Fast Test Reactor (**FTR**) [Li83], and LWR-PV-SDIP [GO89] programs have demonstrated, with great confidence, that without dosimetry measurements, transport calculations can not be used to derive accurate and reliable values of absolute fast flux and fluence for benchmark, test, and/or power reactor applications.

BIASED FLUENCES IN THE CHARPY EMBRITTLEMENT DATABASE -- In a 9th ASTM-EUROPEAN Symposium paper by Worsham [Wo96a] he states:

> *"Currently, calculational results are corrected or adjusted to equal dosimetry measurements and the calculations are used to extrapolate measured vessel fluences."*

In a second Symposium paper [Wo96b] he states:

> *"Corrections or adjustments to equate the calculations and measurements will not provide an acceptable calculational model. The only way to achieve acceptable accuracy and uncertainties in the calculational methodology is to correct any errors and improve the poor approximations. "Measured" vessel fluences that are predicted using erroneous or unreliable calculational models will be erroneous and unreliable. FTI found*

that the previous fluence "measurement predictions in the RG 1.99 [Re88] embrittlement database, are biased. Investigations of (a) the work of Simons, et alia [Si82], Lippincott, et alia [Mc81], and Stallmann, et alia [St86,Mc84], and (b) the reasons for the biased measurements, suggest that the biases are caused by the old FERRET-SAND II technology."

He also states [Wo96b] under "Bias Evaluations - FERRET-SAND and LSL-M2 Comparisons" that:

"If FERRET-SAND produces biased results with fluence values that are too high, as indicated by Equation 5, then the positive bias should be evident when FERRET-SAND results are compared to another least squares adjustment method, such as LSL-M2. Reference [Mc81] provides the comparison of the "PCA Blind Test" results from FERRET-SAND with those from the LSL-M2 predecessor. As would be expected, the FERRET-SAND results showed a positive bias. These results further confirm that the FERRET-SAND adjustments are the reason that the fluences in the embrittlement database are biased. Reference [Mc81] results for the PCA Blind Test were published in 1981. Three years later, after several modifications to the FERRET-SAND procedure, new FERRET-SAND results for the PCA configuration 8/7 were reported, Reference [Mc84]. These results nearly duplicated those reported from LSL-M2. If modifications to least squares technique produce a bias, such as the change between the 1981 and 1984 FERRET-SAND results, the evidence is rather clear the techniques are erroneous."

Worsham's analysis and conclusions are not valid because he did not perform a review and careful study of all of the available PCA/PSF documentation. This is shown by the comparison of PCA [Mc81,Mc84] and PSF [Mc87b] benchmark adjustment code results presented in **Table 3.2.**

The primary reason for the differences between the 1681 and 3318 results is that different input data and assumptions were used for the 1981 and 1984 studies; for example for the 1681 study, Lippincott included Gold's and Roger's differential neutron measurements (proton-recoil spectrometry results) as reported in Sections 3.1 and 3.2. In the 3318 study, these latter data were neglected and not used. Stallmann did not use any of the differential neutron spectrometry data as input to his LSL-M2 analysis. In Section 7.3 of the 1681 report, its stated that:

"The following causes for the discrepancies between the HEDL (FERRET-SANDII) and ORNL (LSL-M2) calculation have been tentatively identified:

1) The HEDL calculations include, in addition to reaction rate data, ^{6}Li-spectrometry and proton-recoil data including that absolute data as reported in Section 3.3.4. The large amount of input information tends to decrease the uncertainties, but may introduce biases if the data are inconsistent.

2) The uncertainties in the foil and fission chamber measurements are also smaller in the HEDL calculations, disregarding the ± 4% core power normalization uncertainty. Photofission corrections were made for the HEDL calculations whereas for the ORNL calculation, the uncertainty for the ^{238}U monitor was increased from 6% to 15%.

TABLE 3.2 COMPARISON OF PCA & PSF BENCHMARK ADJUSTMENT CODE RESULTS

RATIO (NEUTRON FLUX E>1 MEV) FERRET-SANDII / LSL-M2

PRESSURE VESSEL LOCATION	PCA (a) CONFIGURATION NUREG/CR 1681			PCA (b) CONFIGURATION NUREG/CR 3318			PSF (c) CONFIGURATION NUREG/CR 3320 V3
	8/7	12/13	4/12	8/7	12/13	4/12	4/12
SSC-1	-	-	-	-	-	-	0.99
SSC-2	-	-	-	-	-	-	1.02
0-T	-	-	-	-	-	-	1.04
1/4-T	1.08	1.05	-	0.98	1.09	1.07	1.01
1/2-T	1.06	1.06	-	0.98	1.09	1.11	1.01
3/4-T	1.08	1.08	-	1.04	1.10	1.13	-
AVERAGE =	1.07	1.06		1.00	1.09	1.10	1.01

RATIO (NEUTRON FLUX E>1 MEV) SENSAK / LSL-M2

PRESSURE VESSEL LOCATION	PCA CONFIGURATION NUREG/CR 1681			PCA (d) CONFIGURATION NUREG/CR 3318			PSF (e) CONFIGURATION NUREG/CR 3320 V3
	8/7	12/13	4/12	8/7	12/13	4/12	4/12
SSC-1	-	-	-	-	-	-	1.03
SSC-2	-	-	-	-	-	-	0.97
0-T	-	-	-	-	-	-	0.93
1/4-T	-	-	-	1.02	1.07	0.97	0.96
1/2-T	-	-	-	1.03	1.09	0.98	0.96
3/4-T	-	-	-	1.06	1.09	1.08	-
AVERAGE =				1.04	1.08	1.01	0.97

(a) Used Table 7.3.1 results; (b) Used Tables 7.1.2.1, 7.2.1.1 & 7.2.3.1 results; (c) Used Table C1 & Table 4.4.4 results; (d) Same as (b); (e) Same as (c).

3) The HEDL calculations assume much smaller uncertainties for the calculational data (25%) than ORNL (50%). Further, the HEDL calculation has a larger number of groups, which are tied together via short-range correlations."

In Section 4.2.4 of the 1681 report, its stated that:

"That is, the PCA Experiments and Blind Test provides ***necessary****, but not sufficient, validation of the analytical tools and dosimetry methods needed for LWR-PV in vessel neutronic projections (see Section S.1 for a discussion of both necessary and sufficient conditions)."*

See Section 4.2.5 of 1681 on "Recommendations" for additional commentary.

If Worsham had carefully reviewed the PCA reports, he would have realized that he could not make the comparisons between the results of the PCA 1681 and 3318 reports because there were valid reasons (stated above) for the Table 3.2 differences in the PCA results. That is, the preliminary PCA results should not have been used as a basis to support his conclusion that the FERRET-SAND II techniques are erroneous [Wo96b].

To make such comparisons, Worsham should have use the PSF [Mc87b] results generated 3 years later. These results had direct applicability to the validation of the accuracy of PWR and BWR in vessel surveillance capsule and dosimetry methodology. What one finds by comparing the Table 3.2 results of three adjustment codes (FERRET-SANDII, LSL-M2, and SENSAK) is that within the range of the specified uncertainties for the input parameters for the discrete ordinates and Monte Carlo transport calculational and dosimetry measurements, there is ***absolutely no bias*** introduced by the adjustment code methodology. Further, Worsham conclusion that there is a bias in the Reg Guide 1.99, Rev. 2 fluence data base because of the use of FERRET-SANDII derived fluence values in the LWR-PV-SDIP: LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium [Mc87a] is not valid and cannot be justified on a rational scientific basis; see Figure 1, Embrittlement Fluence Factor, in Worsham's paper [Wo97b].

Worsham is a member of the working group on the ANS-19.10 "Fast Neutron Fluence to PWR Reactor Cavities" and would like to maintain the emphasis in DG 1053, as stated, that:

"The prediction of the vessel fluence must be made by an "absolute" fluence calculation in which the transport of the neutron from the core is calculated out to the vessel and cavity, rather than a simple spatial extrapolation of the fluence measurements."

Just prior to the above sentence, it is stated in the Guide that:

"The determination of the pressure vessel fluence is based on both calculations and measurements; the fluence prediction is made with a calculation, and the measurements are used to qualify the calculational methodology.

Because of the importance and the difficulty of these calculations, the method's qualification by comparison to measurements must be made to ensure a reliable and accurate vessel fluence determination. In this qualification, the calculation-to-measurement comparisons are used to identify biases in calculations and to provide reliable estimates of the fluence *uncertainties.*

*When the measurement data are of sufficient quality and quantity that they allow a reliable estimate of the calculational bias (i.e., they represent a statistically significant measurement data base), the comparisons to measurement may be used to **(1)** determine the effect of the various modeling approximations and any calculational bias and, if appropriate, **(2)** modify the calculations by applying a correction to account for bias or by model adjustment or both.*

As an additional qualification, the sensitivity of the calculation to the important input and modeling parameters must be determined and combined with the uncertainties of the input and modeling parameters to provide an independent estimate of the overall calculational uncertainty."

Further, in the Minutes for the June 2, 1997 Orlando ANS-19.10 Meeting [Lo97], it states:

"Worsham stated that he was concerned that the fluences in the Charpy embrittlement database may be biased. He noted that the use of LEPRICON is an interesting concept, but is concerned that these codes do not have a physical basis for the adjustment. In addition, he suggested that any fluence adjustment be made directly to the transport calculations rather than through FERRET and LEPRICON."

It would appear that Worsham supports the NRC and DG 1053 position that the pressure vessel fluence estimates must be based, primarily, on the calculations and not on adjustment code results because:

- Framatome Technology Inc. has direct responsibility to the B&W Owner's Group to provide technical support to its members in the consideration of NRC licensing issues and requirements.

- Appendix H of 10 CFR Part 50 references the ASTM E185 Standard for conducting surveillance tests for many of the detailed requirements of the surveillance program and permits the use of integrated surveillance programs, wherein surveillance program capsules for one reactor are allowed to be irradiated in another host reactor.

Certainly, it could be beneficial for those utilities with B&W reactors (without any surveillance capsules or cavity dosimetry) to be allowed, primarily, to rely on the plant specific calculations for the determination of the best estimate value of the predicated fluence at the pressure vessel inner surface.

In the review and study of the available world wide supporting technical documentation, most countries with NPP now require the combined use of calculations and different types of dosimetry measurements and benchmarking. That is they are using calculated and measured **a)** benchmark mockups **b)** in vessel surveillance capsule dosimetry, **c)** cavity dosimetry, and **d)** vessel and other component metal scraping to obtain the dosimetry measurements needed to verify the accuracy of calculations and the "best estimate" PV fluence value using an appropriate "Adjustment Methodology" as recommended in ASTM Standard E944; see the supporting technical commentary provided in Section 9, "Subsection on 9th ASTM-EWGRD Symposium on Reactor Dosimetry" papers.

It is interesting to observe that in Worsham's statement in the ANS-19.10 meeting minutes he raises the issue that "**The adjustment codes do not have a physical basis for the adjustments.**" These are the same words that Carew and Aronson used in their November 15, 1996 letter to L. Lois on the "Palisades Cycles 1-11 PV and Cavity Fluence Evaluation" [Ca96]. This letter report was attached to the Ref. [Ha96] December 20, 1996 in the Palisades June 26,1996 and December 20, 1996 Information Package.

In Ref. [Ca96], Section IV.3, Carew and Aronson state:

"A major concern with the application of the FERRET adjustment is that, while the adjustment does provide a best-fit of the measured data, the dosimeter cross sections, measured reaction rates and calculated spectrum adjustments are made without any physical basis. This application of the FERRET adjustment methodology to Palisades is presently being evaluated and the results of this evaluation will be reported separately when completed."

To better understand the basis for Worsham's conclusion that least-squares adjustment codes should not be used for obtaining the best-estimate value of the PV fluence, his two 9th Symposium papers [Wo86a,Wo86b] were carefully review and studied to see if there were technical merit; none could be found. This was best shown in Table 3.2 by a comparison of the PCA and PSF benchmark adjustment code results for the LSL-M2 [St86], FERRET-SAND II [Sc79,Mc67,Mc87a], and the SENSAK [Mc79,Mc83] codes. Copies of relevant excerpts from the NUREG/CR 1681, NUREG/CR 3318, and NUREG/CR 3320, Vol. 3 reports that were used to help reach the conclusion that there wasn't any technical merit are attached as Enclosures 5, 6, and 7, respectively. As stated on page 7.3-4 of the 1681 report,

"The information in Table 7.3.2 suggests that the results of the HEDL and ORNL calculations will closely resemble each other as soon as all differences in the input data are eliminated. This needs to be verified by further studies."

"The lack of final values and resolution of discrepancies in some of the measured data, together with the preliminary nature of the ORNL and HEDL analyses, precludes the determination of final recommended values of the exposure parameters and their uncertainties for the PCA. Such values and uncertainties will be derived once a consensus is reached about final input values and uncertainties."

That is, the final values of the FERRET-SAND II/LSL-M2 ratios will be closer to unity. A listing of the causes for the discrepancies between the HEDL and ORNL calculation have been tentatively identified and are listed below:

As stated on page 7.2-1 3-4 of the 1681 report by Lippincott, Stallmann, and Thomas,

"Comparisons of derived exposure parameter values in the block show differences between the three laboratories of up to 12%. No consistent bias between the results exists, when all

the configurations are considered. These differences will have to be investigated and understood to further increase confidence in the least-squares derived uncertainty."

Uncertainties in the exposure parameters also differ between the three laboratories. ORNL has the lowest uncertainty estimates which reflect the application of a more sophisticated approach and/or tighter tolerances on the input spectrum shape. RR&A has the largest range of uncertainty values; for example, for $\phi(E>1)$, the RR&A uncertainties range from 5% to 16% in the block compared to HEDL values of 6% to 9% and ORNL values of 4% to 7%."

"A comparison of the present results with those previously reported by HEDL and ORNL (Table 7.3.1 of Mc81), indicates improved and closer agreement (previous differences ranged as high as 22%). Improved methods and different assumptions have enabled ORNL to reduce their error estimates by a factor of 2 or more. HEDL uncertainty estimates are now higher because the results of each measurement location were handled individually and the proton recoil data were neglected."

It also must be noted that the foil set [^{130}Rh(n,n'), ^{115}In(n.n'), ^{58}Ni(n,p), ^{27}Al(n,α), ^{238}U(n,f) and ^{237}Np] used for the PCA [Mc81,Mc84] studies was different than the [^{54}Fe(n,p), ^{58}Ni(n,p), ^{56}Ti(n,p), ^{63}Cu(n,α), ^{238}U(n,f), and, ^{237}Np(n,f)] set used for the PSF [Mc87b] and PWRs & BWRs dosimetry programs [Mc87a,Si87]. The ^{54}Fe(n,p), ^{58}Ni(n,p), ^{56}Ti(n,p), ^{63}Cu(n,α), ^{238}U(n,f), and, ^{237}Np(n,f) sensors are recommended (in the ASTM standards) for use in PWR and BWR surveillance capsules to measure the flux and fluence E > 1 MeV. These are the same sensors used for the PSF physics-dosimetry-metallurgy experiments and in the Palisades surveillance capsules.

4.0 REVIEW AND STUDY OF THE PALISADES NUCLEAR PLANT'S (PNP) INFORMATION PACKAGE

The reviewer decided that the review and study of the palisades nuclear plant's information package had to be done while keeping in mind **1)** the regulatory aspects and **2)** the technical aspects of Consumers Energy's Palisades Nuclear Plant's Surveillance Program.

Table 3.1 on "**PWR and BWR Surveillance Program Regulatory Instruments**" was prepared to address these and other aspects of the CTS review and study program. Commentary on Table 3.1 and its Attachment is provided in Section 3.0, Subsection "PWR and BWR Surveillance Program Regulatory Instruments." The Table 3.1 Attachment is associated with "Trend Curve Data Development and Testing" and provides relevant background information on FERRET-SAND II surveillance capsule fluence determination and the development and technical basis for RG 1.99, Rev.2.

In Section 1, commentary on highlights of the CTS review and study effort and the work schedule are presented.

5.0 IS OWNER'S METHOD (WESTINGHOUSE METHOD) OF DETERMINING BEST ESTIMATE FLUENCE BY COMBINING TRANSPORT CALCULATION AND DOSIMETRY MEASUREMENTS TECHNICALLY SOUND

This reviewer concluded that this question, designated as "**Question 1,**" had to be answered two ways; from **1)** a regulatory and **2)** a technical perspective.

Table 3.1 on "**PWR and BWR Surveillance Program Regulatory Instruments**" was prepared to address this question. Commentary on Table 3.1 is provided in Section 3.0, Subsection "PWR and BWR Surveillance Program Regulatory Instruments."

The review and study of the information in Table 3.1 and its Table 3.1 Attachment, the relevant documentation listed in Section 2.0, what has been stated in Sections 3 through 9, as well as other considerations, led this reviewer to conclude that the owner's method (Westinghouse Method) of determining "best estimate" fluence by combining transport calculation and dosimetry measurements is both technically and regulatory sound.

The regulatory guidance provided in DG 1053, however, is not consistent with the existing regulatory guidance provided by Federal Law as now specified in 10 CFR Part 50, Appendix G & H, the PTS Rule 10CFR50.61, RG. 1.99, Rev. 2, and the ASTM E706 Master Matrix Set of LWR Surveillance Standards (that by reference to the ASTM E185 Standard in Appendix H has made the entire set of E706 standards a part of Federal Law).

The main inconsistency here is that in RG DG 1053, it is stated:

> *"The determination of the pressure vessel fluence is based on both calculations and measurements; the fluence prediction is made with a calculation, and the measurements are used to qualify the calculational methodology.*

"The prediction of the vessel fluence must be made by an "absolute" fluence calculation in which the transport of the neutron from the core is calculated out to the vessel and cavity, rather than a simple spatial extrapolation of the fluence measurements."

When the measurement data are of sufficient quality and quantity that they allow a reliable estimate of the calculational bias (i.e., they represent a statistically significant measurement data base), the comparisons to measurement may be used to ***(1)*** *determine the effect of the various modeling approximations and any calculational bias and, if appropriate,* ***(2)*** *modify the calculations by applying a correction to account for bias or by model adjustment or both.*

The emphasis in RG DG 1053 for determining the "best estimate" fluence value is on a calculation validated by measurements while in the ASTM standards it is based on using least-squares methodology to generate the "best estimate" fluence value. Direct use of the transport code results with the appropriate bias factors and uncertainties is another acceptable approach."

Since compliance with DG 1053 is not a regulatory requirement of the USNRC, it is suggested that Consumers Energy just continue to follow the regulatory guidance provided in Federal Law as now specified in 10 CFR Part 50, Appendix G & H, the PTS Rule 10CFR50.61, RG 1.99, Rev. 2, and the referenced ASTM E706 Master Matrix Set of LWR Surveillance Standards. That is, follow the existing regulatory guidance that is incorporated and recommended in the Westinghouse Methodology and the referenced ASTM Standards.

Stan Anderson has prepared a draft of Section 5.0 of the new ANS-19.10 "Fast Neutron Fluence to PWR Reactor Cavities" standard. This section is on "Determination of Best Estimate Fluence;" see attached copy, Enclosure 3. This draft is very well written and uses appropriate information extracted from DG 1053, ASTM Standard E706-(IIE2), Guide for Benchmark Testing of LWR Calculations." and ASTM Standard E944 on "Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance."

6.0 IS THE METHOD USED CONSISTENT WITH THE BASIS OF 10 CFR 50.61, THE PTS RULE?

This reviewer concluded that this question, designated as "**Question 2**," also had to be answered two ways; from **1)** a regulatory and **2)** a technical perspective.

Table 3.1 on "**PWR and BWR Surveillance Program Regulatory Instruments**" was also used to address this question.

The review and study of the information in Table 3.1 and its Table 3.1 Attachment, the relevant documentation listed in Section 2, what has been stated in Sections 3 through 9, as well as other considerations, led this reviewer to conclude that the owner's method (Westinghouse Method) of determining the "best estimate" value of fluence by combining transport calculation and dosimetry measurements is consistent with the basis of 10 CFR 50.61, the PTS Rule.

The method is also consistent with the basis of RG 1.99, Rev. 2 because of Consumers Energy (Westinghouse Methodology) use of the FERRET-SAND II physics-dosimetry least squares adjustment methodology recommended in the ASTM E944 Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance. The same methodology used to develop the *NUREG/CR 3319 LWR-PV-SDIP: LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium* [Mc87a,Si87]. [5] That is, it involves the use of the FERRET-SAND Code's adjustment methodology to derive surveillance capsule "best estimate" values of fluence (E>1 MeV) as reported in the Compendium. This "NRC Physics-Dosimetry Compendium's" best estimate fluence values were used as input for Guthrie's derivation of Charpy trend curves based on 177 PWR data points [Gu84]. Randall [Ra86] in turn, then used Guthrie's results together with those of Odette [Od84] to develop Revision 2 of Regulatory Guide 1.99. [Re88]; see Ron McElroy's commentary in Section 3.0, Subsection "Key Issues," on "Regulatory Guide 1.99, Revision 2 Data Base Consistency Issues" and "Consistency of Methodology Issue." Also, see the Table 3.1 Attachment associated with "Trend Curve Data Development and Testing," which includes additional information on the development and technical basis for RG 1.99, Rev.2.

As already stated, the regulatory guidance provided in DG 1053 is inconsistent with the existing regulatory guidance provided in Federal Law as now specified in 10 CFR Part 50, Appendix G & H, the PTS Rule 10CFR50.61, RG 1.99, Rev. 2, and the referenced ASTM E706 Master Matrix Set of LWR Surveillance Standards.

The main inconsistency here is that if the regulatory guidance in RG 1053 were continued to be used to derive "best estimate" fluence values for subsequent PTS screening criteria analyses; the value of the RT_{PTS} estimate will be biased in some undefined way; again, see the Table 3.1 Attachment, Section 3, Subsection on "Key Issues," and Section 9 commentary.

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[5] Also designated as the NRC Physics-Dosimetry Compendium; Section 2.2 of Ref. [Mc86].

Such biases have been minimized in the RG 1.99, Rev. 2 best estimate values of fluence (E> 1 MeV) database because of the use of the ASTM 944 recommended adjustment code physics-dosimetry methodology in the development of the NRC Physics-Dosimetry Compendium [Mc87a]; See the Table 3.1 Attachment, Subsection on NRC Physics-Dosimetry Compendium.

In this regard, PTS Rule 10CFR50.61 requirements state that:

> *"For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{PTS} must use the calculation procedures given in paragraph (c)(1) of this section, except as provided in paragraphs (c)(2) and (c)(3) of this section."*
>
> **Section (c) states that:**
>
> *"Calculation of RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. RT_{PTS} must be evaluated using the same procedures used to calculate RT_{NDT} as indicated in paragraph (c)(1) of this section, and as provided in paragraphs (c)(2) and (c)(3) of this section."*

What does this mean? Answer: It means that if Consumers Energy where to follow the guidance in DG 1053 they would not be in full compliance with Federal Law by not using the recommended physics-dosimetry-metallurgy procedures and methodology as specified in the current version of the ASTM E185 and the other referenced ASTM E706 Master Matrix Standards. In this regard, it is again repeated that the current Code of Federal Regulations guidance is contained in 10 CFR Part 50, Appendix G & H, the PTS Rule 10CFR50.61, RG 1.99, Rev. 2, and the referenced ASTM E706 Master Matrix Standards.

7.0 HAS OWNER'S EXPLANATION OF THE BIAS BETWEEN MEASUREMENT AND CALCULATIONS PROVIDED SUFFICIENT BASIS TO SUPPORT THE MAGNITUDE OF THIS DIFFERENCE?

This reviewer concluded that this question, designated as "**Question 3**," also had to be answered two ways; from **1)** a regulatory and **2)** a technical perspective.

Table 3.1 on "**PWR and BWR Surveillance Program Regulatory Instruments**," and its Table 3.1 Attachment were also used to address this question.

Based on the review and study of all of the material identified in Section 2.0 and what has been stated in Sections 3 through 9, the **answer to this question is a decided yes!**

8.0 HAS OWNER COMMUNICATED ITS POSITION CLEARLY AND IF NOT WHERE WOULD FURTHER EXPLANATION BE USEFUL?

This reviewer concluded that this question, designated as "**Question 4**," also had to be answered two ways; from **1)** a regulatory and **2)** a technical perspective.

Table 3.1 on "**PWR and BWR Surveillance Program Regulatory Instruments**" and its Table 3.1 Attachment were also used to address this question.

Based on the review and study of all of the material identified in Section 2.0 and what has been stated in Sections 3 through 9, the **answer to this question is also a decided yes!**

9.0 FERRET-SAND II METHODOLOGY

REGULATORY REQUIREMENTS AND ASTM STANDARDS -- As discussed in [Mc88], for each LWR nuclear power plant, the physics-dosimetry-metallurgy surveillance program requirements are identified through the **ASTM Standard E185 on "Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706(IF),"** which in turn is identified as a part of the 10 CFR PART 50, Appendix H.

- The requirements for design of the surveillance program are static for each plant. However, the testing and reporting requirements are updated along with technical improvements made to the ASTM E185 Standard.

- **Section 8.2.1 of ASTM E185-94 states that**: *"The neutron fluence rate, neutron energy spectrum, and neutron fluence of the surveillance specimens and the corresponding maximum values for the reactor vessel shall be determined in* ***accordance with the guidelines in ASTM Standard E482 on "Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance" and ASTM Standard E560 on "Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results."***

  Section 8.2.2 states that: *"The specific method of determination shall be documented."*

  Section 8.2.3 states that: *"The Neutron fluence rate and fluence values (E>0.1 and 1 MeV) and dpa rate and dpa values shall be determined and recorded using a calculated spectrum adjusted or validated by dosimetry measurements."*

- **Section 3.1.1 of ASTM Guide E482 states that:**

  "The methodology recommended in this guide specifies criteria for validating computational methods and outlines procedures applicable to pressure vessel related neutronic calculations for test and power reactors.

  Section 3.2 Validation states that:

  Prior to performing transport calculations for a particular facility, the computational methods must be validated by comparing results with measurements made on a benchmark experiment. Criteria for establishing a benchmark for the purpose of validating neutronic methodology should include those set forth in ***ASTM Standard 944 on Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance"*** *as well as those prescribed in* ***Section 3.2.1. Requirements for Benchmarks****.*

  Section 3.4.8.2 states that:

  Use a spectrum adjustment procedure as recommended in Guide E944 using calculated group fluences and dosimetry data with uncertainty estimates to obtain an adjustment *to the calculated group fluences and exposure parameters. Predicted pressure vessel fluences could then incorporate the spectral and normalization data obtained from the adjusted fluences.*

- **Section 4.2.3 of ASTM Practice E560 states that:**

  "Guide E944 should be used to combine the transport calculation with the dosimeter results. The E944 adjustment procedure should be used to indicate whether the dosimeter measurements and associated uncertainties are consistent with the transport calculation and with uncertainties implied from benchmark tests of the transport code (PCA, VENUS, NESDIP, and appropriate Commercial BWR or PWR). Having established the required consistency, the adjusted transport code results may be used to calculate the neutron field at all points in the pressure vessel wall with the uncertainties estimates derived from the application of the adjustment codes. Direct use of the transport code results with the appropriate bias factors and uncertainties is another acceptable approach."

- **Section 3.1 of ASTM Guide E944 states that:**

  "Adjustment methods provide a means for combining the results of neutron transport calculations with neutron dosimetry measurements in order to obtain optimal estimates of neutron damage exposure parameters with assigned uncertainties. The inclusion of measurements reduces the uncertainties for these parameter values and provides a test for the consistency between measurements and calculations and between different measurements. This does not, however, imply that the standards for measurements and calculations of the input data can be lowered; the results of any adjustment procedure can be only as reliable as are the input data."

WESTINGHOUSE METHODOLOGY & TECHNICAL ISSUES -- The Consumers Energy Method (Westinghouse Method) of determining the palisades nuclear plant's best estimate fluence by combining transport calculation and dosimetry measurements is, as described above, in complete compliance with current regulatory requirements and is based on the application of the methodology and procedures recommended in the ASTM E706 Master Matrix Set of LWR Physics-Dosimetry-Metallurgy Surveillance Standards.

Enclosure 2 to this letter report is Section 3 of Consumers Power Company Reactor Vessel Neutron Fluence Measurement Program for Palisades Nuclear Plant - Cycle 1 through 11 (Attachment 2 of Ref. [Sm96]).

In Section 3, it states:

"As noted in Section 1 of this report, the best estimate exposure of the reactor vessel was developed using a combination of absolute plant specific neutron transport calculations and plant specific measurements from the reactor cavity and internal surveillance capsules. In this section, the neutron transport and dosimetry evaluation methodologies are discussed in some detail; and the approach used to combine the calculations and measurements to produce the best estimate vessel exposure is presented."

In discussing the Westinghouse Neutron Transport & Dosimetry Evaluation Methodology used for the PNP, reference is made to 12 ASTM standards,[6] including the **ASTM E706** Master Matrix Standard, the **ASTM E853** "Standard Practice for Analysis and Interpretation of LWR Surveillance Results," the **ASTM E261** "Standard Practice for Determining Neutron flux, Fluence, and Spectra by Radioactivation Techniques," and the **ASTM E262** "Standard Method for Measuring Thermal Neutron Reactions and Fluence Rates by Radioactivation Techniques."

The other eight standards are associated with the guidance for the accurate measurement of the disintegration rates (used to derive the reaction rates) for the dosimetry sensors: **ASTM E263** (Fe54), **ASTM E264** (Ni58), **ASTM E481** (Co & AG), **ASTM E523** (Cu63), **ASTM E526** (Ti46), **ASTM E704** (U238), and **ASTM E705** (Np237). As stated in these standards, general practice indicates that disintegration rates can be determined with a bias of $\pm$ 3% (1σ) and a precision of $\pm$ 1% (1σ) for the non-fission sensors and with a bias of $\pm$ 5% (1σ) and a precision of $\pm$ 1% (1σ) for the fission sensors.

The specific assignment of uncertainties in the measured reaction rates and the input (trial) spectra used in the FERRET evaluations are specified on page 3-15 of Section 3. For reaction rates, a value of 5% was specified. The values specified for flux normalization and fast flux groups are 30%, which clearly shows the expected rather high uncertainty in the DORT absolute calculation of the input values of the shape and magnitude of the neutron flux-spectrum.

The establishment and usage of a dosimetry reaction cross-section and uncertainty file for FERRET-SAND II and other adjustment codes are discussed in the **ASTM E1018** "Standard Guide for Application of ASTM Evaluated Cross Section Data File."[6] It is stated in Section 7.3 of this Guide on Uncertainty File Usage:

"The cross section uncertainty file shall be used as one input to the determination of the overall uncertainties of processed quantities such as fluences and dpa. It is expected that, using least squares adjustment codes such as FERRET, LSL-M2, STAY'SL, or LEPRICON, a good statistical evaluation of the uncertainty of processed quantities can be obtained. The use of validated cross section and uncertainty files will provide the needed confidence to justify usage of derived exposure parameter values and uncertainties for defining neutron-induced material property change limits for LWR nuclear power plants."

For the FERRET-SAND II evaluations, the dosimetry reaction cross-section and uncertainty files were taken from the RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium," July 1994 [Gr93].

For the Np237, U238, Fe54, Ni58, Ti46, and Cu63, Mannhart [Ma89] has measured the spectral average cross sections for each of these reactions in the Cf-252 neutron spectrum and compared the measured values with calculated values using the ENDF/B-V cross section file. The observed C/M ratios were 0.999, 0.970, 1.011, 0.964, 0.935, and 0.955, respectively.

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[6] Annual Book of ASTM Standards, Vol. 12.02, ASTM, Philadelphia, Current Edition.

Table 3 of the most recent revision of the ASTM E261-96 Standard provides an updated listing of these C/M ratios for both the Cf-252 and U-235 fission spectra with the uncertainty in the ratio represented by a sum in quadrature of the experimental and calculated uncertainty. The cross section and spectrum components of the uncertainty are also listed. For Cf-252 the ratios and 1σ uncertainties are 0.981 (9.43%), 0.970 (1.76%), 1.014 (2.65%), 0.981 (2.83%), 0.891 (3.24%), 0.984 (3.72%), respectively. For U-235 the ratios and 1σ uncertainties are 0.990 (11.0%), 0.991 (4.98%), 0.996 (5.91%), 0.974 (7.16%), 0.899 (6.86%), 1.042 (12.9%), respectively. For Cf-252 and U-235, the calculated fission spectrum shape component of the uncertainty for the six reactions range between 0.23 to 1.38% and 4.21 to 6.05%, respectively. For both spectra, the calculated energy dependent spectral averaged cross-section component of the uncertainty for the last five reactions range between 0.53 to 2.85%; the value for Np-237 was ~ 9.3%. These results are based on the most recent work of Griffin et al [Gr93] and on other work reported in seven references dating back ~ 22 years to the November 1976 Vienna IAEA Consultants Meeting on Integral Cross-Section Measurements in Standard Neutron Fields [Fa78] and the USAEC's Interlaboratory LMFBR Reaction Rate (ILRR) Program [Mc75,Mc75a,Mc77]. The cross section and covariance matrices used for Table 3 evaluations are consistent with the source detailed in Ref. [Gr93].

Only the Np237, U238, Fe54, and Ni58 reactions have a significant effect on the adjusted value of the fluence (E> 1 MeV) because of the high energy response (E> ~ 4 MeV) for the Ti46 and Cu63 sensors. There are not enough neutrons above ~ 4 MeV to significantly influence the final FERRET-SAND II adjusted value of the fluence for neutrons above 1 MeV.

This review of uncertainties in the sensor reaction data bases demonstrates that the statements made by Carew and Aronson in Section V1.2 of Ref. [Ca96], below, that there are "several other possible explanations" do not have technical merit!

*"In the CPC/W analysis this difference is assumed to be due to a spectrum-dependent error in the DORT calculations which results in an exact calculation above E> 4.0 and an over prediction for E< 4.0 MeV.................. Based on this assumption, a 12% M/C fluence reduction (i.e., not including the additional FERRET reduction) is applied to the DORT E> 1.0 MeV fluence prediction. The application of this M/C spectrum-dependent correction is illustrated in Figure 6. While this conclusion may be correct, there are several other possible explanations for the observed 1.00/0.86 difference between the high/low-energy M/C biases that would not require this reduction in the DORT calculated fluence. These include: **1)** the use of erroneously low dosimeter cross sections for Fe-54 and Ni-58 in the interpretation of the measurements and/or **2)** errors in the Fe-54 and Ni-58 measurements."*

That is, explanations 1) and 2) above cannot be justified on a technical basis and are not possible explanations for the observed biases between the Fe54 & Ni58 and Ti46 & Cu63 results. The reason for the biases is that some of the input information for the calculation is incorrect, such as the magnitude of the fission source term and the high energy (E> 4 MeV) part of the input form of the composite U235, U238, and Pu239 fission spectrum. The FERRET-SAND II result for the adjusted form of the DORT calculated input flux-spectrum is correct and cannot be challenged on the basis of errors in the sensor measured reaction rates and ENDF/B-V or VI energy dependent evaluated dosimetry and uncertainty data files!

This is further confirmed by a careful study of the Table 3.2 results for the PSF where the average FERRET-SAND II/LSL-M2 ratios and SENSAK/LSL-M2 ratios for the six PWR pressure vessel mockup locations for the PSF 4/12 configuration are 1.01 and 0.97, respectively. As stated in Section 3.0, Subsection on Biased Fluences in the Charpy Embrittlement Database, of this letter report:

*"What one finds by comparing the Table 3.2 results of three adjustment codes (FERRET-SAND II, LSL-M2, and SENSAK) is that within the range of the specified uncertainties for the input parameters for the discrete ordinates and Monte Carlo transport calculational and dosimetry measurements, there is **absolutely no bias** introduced by the adjustment code methodology."*

In Section V1.3 of Ref. [Ca96], it is stated that:

"A major concern with the application of the FERRET adjustment is that, while the adjustment does provide a best-fit of the measured data, the dosimeter cross sections, measured reaction rates and calculated spectrum adjustments are made without any physical basis."

In response to this statement, it can be said that the statement is not relevant because the use of a least-squares adjustment code allows the study of biased data. That is, the analyst must not only apply physically based physics and reasoning but must carefully examine the data to determine if it is of adequate quality, which has been done for both the PNP and PSF FERRET-SAND II Code analyses.

The FERRET-SAND II Adjustment Code Methodology is well documented in the Section 3 (Enclosure 2) report and elsewhere [Li83,Li86,Li94,Mc87a,Wo81]. Further, the methodology is based on the guidelines provided in a number of the applicable ASTM standards; see Table 3.1. The FERRET-SAND II, LSL-M2, SENSAK, LEPRICON and other adjustment code methodology is well established, has been validated, and has been accepted nationally and internationally for deriving "best estimate" fluence values for test and power reactor applications; see ASTM Standard E944 [6] and Figure 2 and Table 1 of the next Subsection's Paper 1, "Dosimetry in Support of the European Network AMES."

It is very discouraging, therefore, to have the NRC staff say in their December 20, 1996 letter to Thomas Bordine, Manager of Licensing, Palisades Nuclear Plant [Bo96], that:

"Our review finds that: 1) the fluence reductions based on physical quantities to be acceptable, because they are based on directly measured parameter, 2) the 12% bias is not acceptable because the measurements are inconsistent and statistically incompatible, and 3) the spectral adjustments of 5% are not acceptable because they represent an intuitive averaging and do not evoke any physical principle. The spectral adjustments (5%) are still under evaluation; however, it does not seem likely that the method is adequately justified."

Based on the ASTM E944 Standard's recommended and validated use of FERRET-SAND II, LSL-M2, SENSAK, LEPRICON and other adjustment codes referenced in E944 (as well as the commentary and results presented in Papers 1, 2, 3, 4, and 5 of the next Subsection on the 9th ASTM-EUROPEAN Symposium on Reactor Dosimetry), there is no scientific technical merit or justification for making such a statement!

or justification for making such a statement!

In [Ca96], and as previously quoted, Carew and Aronson state in **Section VI.1, Best-Estimate Fluence Determination**:

" ....Based on these measurement-to-calculation (M/C) comparisons of the dosimeter reaction rates, a M/C bias of 12(± 7)% is determined. This M/C bias is then adjusted using a least-squares adjustment technique to account for uncertainties in the measurement and calculations. In the case of Palisades this adjustment increases the M/C bias from 12% to 17%, and implies the calculations are over predicting the fluence by 17%. The determination of the M/C bias and the adjustment method are discussed in the following sections."

In **Section VI.2, Fluence Calculation-to-Measurement Bias**, they state:

"From Figure 4, it is seen that the in-vessel M/C bias is 1.00 ± 0.03 for the dosimeters with thresholds E> 4.0 MeV, and 0.86 ± 0.02 for the dosimeters with thresholds E< 4.0 MeV. In the CPC/W analysis this difference is assumed to be due to a spectrum-dependent error in the DORT calculations which results in an exact calculation above E> 4.0 and an over prediction for E< 4.0 MeV. Based on this assumption, a 12% M/C fluence reduction (i.e., not including the additional FERRET reduction) is applied to the DORT E> 1.0 MeV fluence prediction. The application of this M/C spectrum-dependent correction is illustrated in Figure 6. While this conclusion may be correct, there are several other possible explanations for the observed 1.00/0.86 difference between the high/low-energy M/C biases that would not require this reduction in the DORT calculated fluence. These include: 1) the use of erroneously low dosimeter cross sections for Fe-54 and Ni-58 in the interpretation of the measurements and/or 2) errors in the Fe-54 and Ni-58 measurements. (The number of U-238 and NP-237 dosimeters that are included in the in-vessel M/C bias is small and these measurements are subject to relatively large uncertainties."

In **Section VI.3, Least-Squares Fluence Adjustment**, they state:

"A major concern with the application of the FERRET adjustment is that, while the adjustment does provide a best-fit of the measured data, the dosimeter cross sections, measured reaction rates and calculated spectrum adjustments are made without any physical basis. This application of the FERRET adjustment methodology to Palisades is presently being evaluated and the results of this evaluation will be reported separately when completed."

NINTH ASTM-EWGRD SYMPOSIUM ON REACTOR DOSIMETRY -- In order to provide additional supporting technical documentation and commentary related to the technical issues raised in the above statements, a preliminary review and study was completed of approximately ~40 of the most relevant papers of the ~80 papers distributed at the 9th ASTM-EWGRD Symposium on Reactor Dosimetry, held in Prague, Czech Republic, September 2-6, 1996.

This review and study was only partially completed, but it did provide additional information that strongly supports the use and accuracy of the application of the ASTM and Westinghouse

Methodology and use of the FERRET-SAND II or other adjustment codes. [7] Of particular importance here, is the supporting technical commentary (related to selected methodology, standardization and technical issues) provided in five of the Prague papers. The first paper provides commentary related to the European Network on Ageing Materials Evaluation and Studies (**AMES**) that was set up in 1993 to bring together the organizations in Europe that have the main capabilities on reactor pressure vessel materials assessment and research. The second paper provides commentary related to the European Organization for Economic Cooperation & Development (**OECD**), Nuclear Energy Agency (**NEA**), Nuclear Science Committee (**NSC**) study on Issues of Dosimetry Fluence Computations.

The European Network on AMES projects and activities and the OECD-NEA-NSC studies are providing a very important continuation and expansion of:

1) The previous NRC supported LWR-PV-SDIP related activities at HEDL, ORNL, NBS (NIST), MEA, and UCSB and those supported by CEN/SCK (Mol), by EPRI/Utilities at B&W, CE, GE & W, by SwRI & BMI, by several UK Laboratories (Harwell; Rolls-Royce & Associates; Winfrith), and by KFA (Julich, Germany), CEA/CEN (Saclay, France), GKSS & IKE (Germany), EIR & HSK (Switzerland), the Joint Japanese Special Working Group (UT, JAERI, TC, MHI, TU, Hatachi, JSW and MSPI); and

2) Present and past ASTM Standards Technology Development, Transfer, and Training (**STDTT**) activities and ASTM's sponsorship and co-sponsorship, respectively, of the series of International ASTM Symposia on the Effects of Radiation on Materials and the ASTM-EWGRD Symposia on Reactor Dosimetry.

__Paper 1__: **"DOSIMETRY IN SUPPORT OF THE EUROPEAN NETWORK AMES"**

As stated by Ballesteros, Debarberis, and Voorbraak in their Prague paper [Ba96]:

"There is a need to coordinate activities to ensure maintenance of capabilities and facilities, and also to harmonize activities with the objective of common European standards. A European network, AMES (Ageing Materials Evaluation and Studies), was set up in 1993 to bring together the organizations in Europe that have the main capabilities on reactor pressure vessel (RPV) materials assessment and research *[Ba96a,Br95,Es95].*

Several tasks, in the dosimetry field, have been planned for the AMES projects and activities. In addition, a study is being carried out with the following objectives:

- *review of the current situation for different considered reactor types (including WWER reactors)*
- *summarize the existing reference documentation, methodology and techniques*

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[7] Related to this review and study, this letter/report's Acronyms listing includes some of the acronyms used in ~ 40 technical papers.

- *analyze key factors for European harmonization*
- *validation and benchmarking of dosimetry practices*
- *establish a set of recommendations for the AMES projects*
- *post-mortem dosimetry validation*
- *limitations, uncertainty study and possible improvements*

A description of the dosimetry activities in support of the AMES European network will be presented. Conclusions and recommendations are also of interest for the dosimetry community, and will contribute to a closer working relationship between specialists in both materials and dosimetry."

The following additional commentary was taken from the Ref. [Ba96] paper:

WORKING METHOD

Regarding dosimetry, three options were evaluated as working methods to reach the objectives of this Task Group:

Option 1: *Incorporate dosimetry in the materials research proposals.*

Option 2: *Dosimetry proposals are independent and complement materials research proposals.*

Option 3: *Combine Option 1 and Option 2. This last option has been considered the most adequate for the AMES projects.*

As an example of application of Option 1, a specific task on retrospective (a posteriori) dosimetry has been included in a project proposal relative to the analysis of 20 trepans available from the core baffle of a Spanish NPP. In this project proposal three different disciplines are involved, neutron metrology (experimental and calculations), material research and fracture mechanics.

Relating to Option 2, a project proposal has been elaborated and it includes:

1. *Neutron calculations concerning all different European reactors types (PWR, BWR, Magnox, WWER and materials test reactors).*

2. *A benchmark with VENUS data. SCK/CEN will supply all necessary data and will coordinate the benchmark activities.*

3. *Analysis of the impact of uncertainties in fluence calculation procedures, for specific reactors, on the brittle behavior of RPV materials, according to the embrittlement trend curves used in different countries.*

4. *Compile a conversion table between typical reactor locations (surveillance position, peak flux positions at inner wall, 1/T, 3/4T, etc.) for different reactor types in as many indices as possible. Different indexes are used for neutron damage. Normally* $\sim > 0.1$ *MeV and* $\sim > 1.0$ *MeV for Western materials, and* $\sim > 0.5$ *MeV for Russian materials.*

Another indexation used mainly for Magnox reactors is the number of displacements per atom DPA. A direct comparison of the above mentioned quantities is not possible in general. The neutron spectra are different for the different cases as well as the material nuclear libraries. Therefore, it is recommendable the generation of a qualified conversion table of all used damage indexes in order to compare available data and studies.

CURRENT ACTIONS

A first action, performed before the formal constitution of the Task Group, was to prepare a State-of-the Art report on Dosimetry [Ba96a]. This report, financed by EC-DG XI and available for July 1996, shows the international situation on dosimetry and neutron calculations for reactor pressure vessel surveillance. It establishes general recommendations, and identifies technical areas where more investigation is needed. Its main usefulness is to be a reference document to take decisions on further R&D. Figure 2 depicts the detailing plan for the study. (Note, to the PCA/PSF, NESDIP, and VENUS benchmarks, they have added LR-O [Os96, Ho96, Ho96a].) Table 1 and 2 have been included in the final report and show examples of neutron transport methodologies and existing ENDF/B and derived libraries, respectively.

Some other relevant actions of the Task Group are described below.

1. *It was noted the necessity to include a very well calibrated dosimeter, or a set covering the full range of energies, in all AMES irradiations in order to reduce uncertainties that could affect neutron damage assessment. Some desirable characteristics of this State-of-Art dosimeter are:*

   - *Easily calibrated.*
   - *Stable.*
   - *Relatively insensitive to extremes of environmental conditions.*
   - *Correctable systematic errors.*
   - *Produced in reproducible lots.*
   - *Small dimensions compared to distances over which neutron flux gradients become significant. This is of special interest for AMES irradiations in WWER reactors.*

   It was agreed to develop the specification for this correlation dosimeter.

2. *Ex-vessel dosimetry and retrospective (a posteriori; e.g., vessel wall scrapings.) dosimetry are of great interest for AMES. Several specific tasks are being planned for the materials research project proposals."*

In regard to the Figure 2 LR-O Benchmark, Osmera et al. [Os96] have stated:

"The experimental and theoretical investigations of the reactor dosimetry problems started in NRI and Skoda 20 years ago. Due to the unsatisfactory reliability of the calculations and input data libraries the programme of the LR-O Mockup-Ups has been launched." (See Paper 3 for detailed commentary on the results of the "LR-O Benchmark experimental and theoretical PWR studies on VVER-1000 reactor dosimetry."

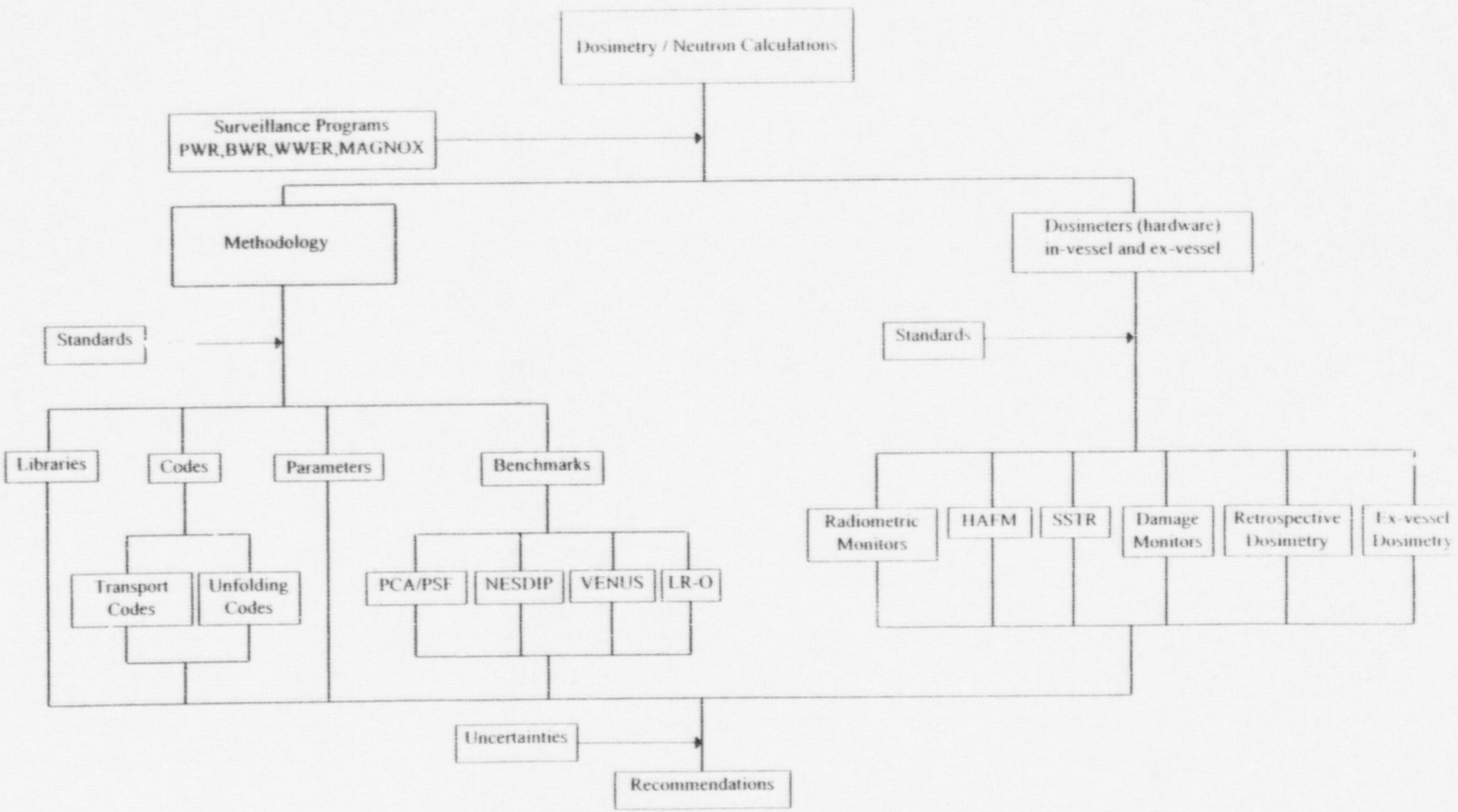

FIGURA 2: DETAILING PLAN FOR THE STUDY

TABLE 1: EXAMPLES OF NEUTRON TRANSPORT METHODOLOGIES

REACTOR	TYPE	NEUTRON CROSS-SECTIONS	NEUTRON CODE	DOSIMETRY CROSS-SECTIONS	ADJUSTMENT CODE	REF. (1)
LOVIISA	WWER-440	BUGLE-80	DOT 3.5	IRDF-90		3 p45
MAINE YANKE	W	BUGLE-80 SAILOR	DORT (S8P3)	ELSXIR		3 p650
TIHANGE-2	W	ELXSIR	DOT 4.3 (S8P3)	ELXSIR	LEPRICON	3 p65
KOEBERG	W	VITAMIN-C	DOT 3.5	IRDF-90		3 p155
DOEL 1-2	W	VITAMIN-C	DOT 3.5 (S8P3, 17g)			1 p17
PAKS	WWER-40			IRDF-85		1 p105
KRSKO	W	DLC-2D	DOT 3.5 (S6P3)	IRDF-82	STAY'SL	1 p115
		SAILOR	DOT 4.2 (S8P3, 27g)	IRDF-85	LSL-M2	2 p63
LR-O	WWER-440 mock-up	VITAMIN-C	DOT 3.5 (S8P3)			1 p130
McGUIRE ANO-I ANO-II	W B&W CE	XSDRNPM	DOT 4.2	ENDF/B V		1 p139
B&W Owner Gr.	B&W		DOT 4			1 p155
KRB-A (2)	BWR	ENDF/B IV	DOT 4.2			1 p165
H.B.ROBINSON	W	SAILOR	DOT 3	ENDF/B V	FERRET	1 p147
		ELXSIR	DOT 4.3	ELXSIR	LEPRICON	1 p405
Saint Laurent B1	W		TRIPOLI		STAY'SL	2 p73

(1) Ref = Reference:, n pjj = reference n, page jj

Ref.1 = 6° ASTM-Euratom Symposium, June 1987, ASTM STP 1001

Ref.2 = 7° ASTM-Euratom Symposium, August 1990, Proceedings. Kluwer Academic Publishers

Ref.3 = 8° ASTM-Euratom Symposium, September 1993, ASTM STP 1228

(2) Decommissioned

TABLE 2: ENDF/B AND DERIVED LIBRARIES

YEAR	ORIGINAL LIBRARY	DERIVED LIBRARY	TYPE	NUMBER OF GROUPS	REFERENCE	PUB
1972	ENDF/B III	DLC-2/100G	X	100	DLC-2	
1978	ENDF/B IV	VITAMIN-C	X	171	DLC-4	(3)
1984	ENDF/B V	VITAMIN-E	X	174	DLC-113	
1993	ENDF/B VI	VITAMIN-B6	X	199		
1980	ENDF/B IV	BUGLE-80	X	47	DLC-75	
1983	ENDF/B IV	SAILOR	X	47	DLC-76	
1984	ENDF/B IV + V	ELXSIR	X,D	56		(1)
1993	ENDF/B VI	BUGLE-93	X	47		
1982	ENDF/B III	IRDF-82	D	620	DLC-94	(4)
1984	ENDF/B IV + V	ELXSIR	X,D	56		(1)
1985	ENDF/B IV	IRDF-85	D	620		(5)
1990	ENDF/B VI	IRDF-90(6)	D	620	IAEA0867	(2)

(0) X = neutron cross- sections; D = dosimetry cross-sections

(1) EPRI NP-3654 (1984)

(2) IAEA-NDS-141 (OCT/93)

(3) ORNL-RSIC-37

(4) IAEA-NDS-41/R (1982)

(5) IAEA-NDS-41 (OCT/93)

(6) IRDF-90/G V.2, version October 1993

Note: The inelastic scattering cross section of iron was modified to a large extent in the ENDF/B V mod. 3 (1986). This new cross-section is not included in SAILOR. On the contrary, it is included in the last versions of ELXSIR.

Paper 2: "ORGANIZATION FOR ECONOMIC COOPERATION & DEVELOPMENT, NUCLEAR ENERGY AGENCY (NEA), NUCLEAR SCIENCE COMMITTEE (NSC) STUDY: ISSUES OF DOSIMETRY FLUENCE COMPUTATIONS"

The following commentary was taken from the Rulko, Kodeli and Sartori Prague paper [Ru96]:

"In order to determine metal damage due to neutron (and gamma if significant) irradiation fluence and spectra have to be determined at points of interest in a reactor from transport calculations and experimental measurements. This fluence is then used in some damage model to infer the damage of metal. In addition, the coupling between the experimental results and transport calculations takes place since fluence is derived from the measurements and/or the calculated neutron spectrum and dosimeter reaction rates are derived from the calculations and measured decay rates using the flux history and transport results.

The measurements are used to validate the calculational results. In addition, the intimate interdependence of calculations and measurements is often used to adjust the calculated spectra with measurements via least-squares fitting. Usually, uncertainties in measured values are much lower than those in calculations and hence adjusting permits to combine the measurements and the calculations in a way to provide the adjusted fluence with the reduced uncertainties. Typical recent results show uncertainties of 15 to 20 percent for unadjusted methods and results within the measurement uncertainties of 5 to 10 percent for adjusted methods. *Opinions vary if the adjustment of calculated spectra using experimental data is necessary or not. If the object of a given dosimetry study is a validation of transport methods then different unadjusted results of calculations should be compared against measurements to deduce the sources of computational errors due to methods used, cross-section data, and modelling assumptions.*

COMPUTATIONAL METHODS AND UNCERTAINTIES

*The computational schemes presently used fall into two major categories of discrete ordinates methods and Monte Carlo methods both combined with sensitivity and uncertainty analysis whose application has proved necessary to establish reliable safety margins for target quantities. Sensitivity analysis is the most effective measure of predicting what model and data improvements are most effective in predicting target quantities. Uncertainty analysis is essential in establishing the level of confidence one can have in the results of calculations. There are a number of uncertainties associated with the computations using transport codes. The **first set** of uncertainties results from numerical approximations such as: the order of quadrature used in S_N calculations, the scattering cross-section expansion, choice of meshes in the model, energy group structure, statistical convergence criteria, etc. The **second set** of uncertainties results from modelling approximations such as: surveillance capsule placement, RPV thickness variations, cavity streaming effects, 3-D flux synthesis, peripheral subassembly source distribution, dimension and material uncertainties. The **third set** of uncertainties results from uncertainties in nuclear data such as: cross-sections, dosimeter cross-sections, U235 fission spectrum above 6 MeV.*

A major effort has been devoted to the analysis of cross-section data uncertainties and derived quantities such as displacement per atom (DPA) cross-sections, kinetic energy release in materials (kerma), and gas production data. The full uncertainty analysis requires:

1) *Least-squares methods for estimating the values of cross-sections and their covariances,*

2) *processing of cross-sections and covariances into a multigoup form,*

3) *sensitivity and uncertainty analysis of target quantities (e.g., fluence and doses), and*

4) *a consistency analysis of the experimental and calculated data and improvement through adjustment."*

Paper 3: "EXPERIMENTAL AND THEORETICAL STUDIES ON VVER-1000 REACTOR DOSIMETRY"

The following commentary was taken from the Osmera et al. Prague paper [Os96]:

*"**INTRODUCTION -- The experimental and theoretical investigations of the reactor dosimetry problems started in NRI and Skoda 20 years ago. Due to the unsatisfactory reliability of the calculations and input data libraries the programme of the LR-O mockup-ups has been launched.***

Several VVER-440 mockups with standard and reduced cores have been investigated. The mockup experimental data have been used in evaluation of the neutron monitors in surveillance specimens programme [HO96] and ex-vessel position measurements [Ho96a].

.............The neutron spectra have been measured in the horizonal central plane along the axis of the mock-up symmetry with the proton recoil spectrometers. The power distribution has been checked by the gamma scanning of the fuel pins. The measuring program emphasizes the RPV critical location and the monitoring ones. The 1D and 2D (ANISN, DORT) transport calculations with the SAILOR and BUGLE-93 libraries have been performed.

In the Skoda design of the surveillance program the specimens are placed in the rectangular boxes fixed at the RPV inner wall in the position of the maximum of the fast flux neutron density and the acceleration (lead) factor is about 1.9 for the fast fluence above 0.5 MeV. The specimens boxes are equipped with Fe, Nb, Co, Cu, Np237 and U238 detectors and Cu and Fe wires for space dependent distribution measurement. ....

***EXPERIMENTAL PROGRAMME & EXPERIMENTAL TECHNIQUES** -- ....The LR-O low power limit the measurement with activation detectors, but the differential energy neutron spectrum is a better test of calculation model and methodology than a set of reaction rates usually measured in many benchmarks.*

The radial-azimuthal core power distribution use to be checked by means of the gamma scanning of a reasonable set of fuel pins (several hundreds for the radial and several tents for the azimuthal distribution). The radial and azimuthal distributions of the fast flux in the vicinity of the core or in the central tubes of the fuel assemblies could be measured with the indium foils, in the positions from the core the proton recoil detectors could be used for the distribution measurements.

CALCULATIONS -- *The transport codes ANISN, DORT, TORT and the libraries VITAMIN-C, SAILOR, BUGLE-93 have been implemented. The core distribution was calculated with MOBY DICK, 2D, diffusion, multigroup, pin-to-pin code. Some distributions were calculated using the "influence functions." Using the point kernel method the contributions of 20 -50 layers over the height of the core to the chosen point are calculated. The effective cross sections are evaluated via 1D transport calculations. The influence function calculation (integral of Green functions over a defined area of the core) follows the MOBY DICK results. Using this system it is possible to calculate the time dependent integral (above defined energy) fast neutron flux density distribution in the position of a neutron monitor.*

DISCUSSION OF RESULTS -- *The neutron spectra were measured at the outer surface of the barrel, at the PV inner surface, and the PV outer surface. The experimental spectra were transformed into the SAILOR (BUGLE) group format, and the broad group approximation (0.1, 0.2, 0.5, 1.0, 2.0, 3.0, 7.0 and 10.0 MeV) which were used for the comparison with the calculation. Because the absolute power in LR-O cannot be determined the measurements are relative and a monitoring system covering several orders in the neutron flux density is used. Comparing the shapes of a spectrum the measured and calculated spectra were normalized to integral flux in the energy range 1 to 10 MeV or 0.1 to 10 MeV. ....*

The spectra measured by both spectrometers were identical in the frame of usual uncertainties known from the measurement in reference fields. The proton recoil spectrometers' experimental (higher upper energy limit) and 2D-DORT, BUGLE-93 Skoda calculation were used for the comparison. The sensitivity of the results to the substitution of SAILOR BY BUGLE-93, P7, was studied with 1D (ANISN) calculation. ***The differences (improvements) were remarkable. Nevertheless the discrepancy of the 2D-DORT, BUGLE-93, calculation and experiment is substantial for the evaluation of the reliability of the RPV exposure calculation.***

The calculation mostly underestimated the fast fluxes. [8] *Comparing the calculated and measured space energy distributions and space energy indices, broad groups and integral fluxes above defined energy, it could be stated that the tendency in Fig. 3 (Normalized to integral flux in the energy interval 1-10 MeV at the barrel outer surface position) is similar as in Fig. 2 (Normalized to integral flux in the energy interval 0.1-10 MeV at the barrel outer surface, PV inner surface, and PV outer surface positions) except at the PV outer surface, the disagreements in attenuations of the integral fluxes over the water layer and the RPV (Fig. 4 Measured and Calculated Indices/Comparison of Integral Fluxes) corresponds to the ones in spectra. The results of calculation with the "influence functions" are in Fig. 3 marked as 3D-INCALC.* ***The disagreement of calculation and experiment is probably caused by the used group library. In several studies the sensitivity of the results to the type of the library are presented [Ha96a,Ch92]. The Monte Carlo calculations of similar problems [Osxx,Woxx] show substantially better consistency with experiment."***

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[8] This is the same trend observed for the PCA Experiments and Blind Test. That is, compared with the measurements, most calculations showed a trend towards under-estimating the fluxes (E > 0.1 and 1.0 MeV) by 5% to 25%, and even more at higher neutron energies and deeper steel penetrations [Mc81,Mc84].

Paper 4: "NEUTRON DOSIMETRY IN EXTENDED SURVEILLANCE PROGRAM ON 4TH UNIT OF NPP DUKOVANY"

As stated by Hogel et al. in their Prague paper [Ho96]:

*"**INTRODUCTION** -- The Standard Surveillance Program for the reactors VVER 440/213 of Dukovany NPP demonstrated a number of shortcomings. As to the neutron fluence monitoring the surveillance containers (capsules) were equipped with only a restricted number of activation monitors - 54Fe, 63Cu, and 93Nb. Moreover, the surveillance container chains contained only several sets of monitors and that is why the detailed description of the vertical neutron fluence distribution was not possible............ Therefore it was not possible to determine reliably the neutron fluence in the individual surveillance specimens. As well the neutron fluence on the reactor vessel could be determined with great uncertainty. Uncertainty in the individual surveillance specimen neutron fluence led to problems with the application of the mechanical test results to the RPV state and life time assessment.*

***DETECTORS FOR THE ESP** -- The Extended Surveillance Program (**ESP**) was designed to eliminate all the above described shortcomings.......The ESP included two chains of 20 surveillance containers each. ....The chains were irradiated in the NPP Dukovany Unit 4, Cycle 7.*

Containers were equipped with either Fe and/or Nb wires for neutron fluence distribution assessment both inside the container and along the whole chain. Three types of wires were used:

- *O-wire, a container perimeter neutron fluence distribution monitor, located above the surveillance specimens (in several containers).*
- *I-wire, a container height neutron fluence distribution monitor, located along the surveillance specimens (in all containers).*
- *H-wire, a detailed vertical neutron fluence distribution monitor, inserted directly into the specimen notch (in several containers).*

Specified surveillance containers contained additional spectrometric sets of neutron fluence monitors Fe, Cu, Ni, Nb, alloy Al-Co and fission detectors 238U and 237Np..................... In the second chain the spectrometric sets were equipped with isotopically enriched monitors 54Fe, 63Cu, 93Nb, and 59Co separated by natural pellets of Ti. ....

***ACTIVITY MEASUREMENTS** -- were performed in the two independent laboratories - NRI Rez and SKODA Nuclear Machinery. Both the laboratories used HPGe spectrometer for the standard measurements and HPGe spectrometer with Be window for the Nb X-ray measurements.*

Before measurement the O-wires were divided into 6-8 approximately equal pieces and I-wires into 4-5 pieces depending on their length. By this way the container perimeter and height *neutron fluence distribution was obtained.*

Local changes of the neutron fluence rate along the core height depending mostly on the fuel burn-up during the operation cycle were also taken into account. Correction to the container orientation was made in order to get the activities in the container axis. Nb activities were corrected for the X-ray self-absorption and for the X-ray fluorescence caused by decay of both 94Nb and 182Ta impurities.

***RESULTS AND DISCUSSION** -- Examples of the container height neutron fluence distributions (I-wires) are presented.......... It can be seen that the height distribution changes from 0.8 to 1.2 relative to the mean value in the first container placed at the top chain position. In the containers from the middle part of the chain .... the vertical fluence distribution is close to the container mean value. ....*

A detailed relative neutron fluence distribution in the container .... equipped with both H-wires and O-wires and spectrometric detector set is shown in ....The value of the neutron fluence in the specimen notch varies from 0.83 to 1.16 relative to the spectrometric set position.

Using the height neutron fluence distribution determined from the O-wires the reaction rates for the spectrometric sets were corrected, the results are presented in ....

The reaction rates were adjusted to an a priori neutron spectrum which had been measured by Holman [Ho91], Jansky and Marek [Bu86] on the VVER-440 mock-up using differential proton recoil methods.

The SAND II adjustment procedure was used with IRDF-90 cross section library. Adjusted neutron fluences after one iteration are given in Tables 3 and 4 for the chain No. 1 and 2, respectively. From this process the 238U monitors had to be excluded as they showed inconsistency with the a priori spectrum. Probable reason is the content of 235U was too high for this type of neutron spectrum. The standard deviation of the measured and calculated activities varies from 1.9% to 6.1 which shows good consistency between neutron dosimetry data and the a priori neutron spectrum. [9] *..........*

***CONCLUSIONS** -- From the results presented above the following conclusions can be stated:*

- *The perimeter distribution measurement (O-wire) is necessary so that the detailed description of the neutron fluence field inside the container can be assessed and the vertical neutron fluence distribution along the chain can be estimated.*

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[9] A major problem with using discrete ordinates transport codes is that the group structure is not fine enough to properly handle the effect of fine structure in the actual neutron spectrum and transport code and dosimetry cross sections. Using the higher resolution measured proton recoil spectrum, the 640 group SAND II dosimetry cross sections, and the SAND II iterative (or FERRET-SANDII least-squares) adjustment code eliminates the need for the use of processed broad group averaged cross sections. This problem is also eliminated or minimized by the use of Monte Carlo transport codes with high group structures up to ~ 8000.

- *Only in the few bottom and top containers the vertical neutron fluence distribution in the container should be measured using the H-wires.*

- *Spectrometric sets should be equipped only with Fe, Cu, Ni, Nb, Ti, and Co. The usage of the fission detector 237Np is not necessary as we are able to measure and correct Nb monitors. The 238 monitors are not suitable for this type of neutron spectrum. The content of 235U is too high and its influence cannot be eliminated by a Gd cover. To ensure possibility of Nb correction the Nb monitors should be used both bare and covered by Gd."*

With reference to the commentary presented in Paper 3, Osmera et al. have stated:

"The proton recoil spectrometers' experimental (higher upper energy limit) and 2D-DORT, BUGLE-93 Skoda calculation were used for the comparison. The sensitivity of the results to the substitution of SAILOR BY BUGLE-93, P7, was studied with 1D (ANISN) calculation. The differences (improvements) were remarkable. Nevertheless the discrepancy of the 2D-DORT, BUGLE-93, calculation and experiment is substantial for the evaluation of the reliability of the RPV exposure calculation. ....

The calculation mostly underestimated the fast fluxes. ....

The disagreement of calculation and experiment is probably caused by the used group library. In several studies the sensitivity of the results to the type of the library are presented [Ha96a,Ch92]. The Monte Carlo calculations of similar problems [Osxx,Woxx] show substantially better consistency with experiment."

What the above says is that in the Czeck Republic, by appropriate LR-O Benchmark VVER-440 mockup studies, the NRI-Rez and NPP Dukovany staffs have determined that they <u>cannot</u> rely on absolute unadjusted (by dosimetry measurements) transport calculations to determine reliable surveillance capsule specimen and RPV end-of-life neutron fluence exposure values.

That is, and as Hogel et al. have already stated:

"Therefore it was not possible to determine reliably the neutron fluence in the individual surveillance specimens. As well the neutron fluence on the reactor vessel could be determined with great uncertainty. Uncertainty in the individual surveillance specimen neutron fluence led to problems with the application of the mechanical test results to the RPV state and life time assessment."

Table 3. Fast neutron fluence ($1/cm^2$) in the container axis at various elevation above core midplane, chain No.1.

Elevat. [m]	SD %	0 0.1eV	0 0.398eV	0.398eV 10keV	10keV 0.1MeV	0.1MeV 20MeV	0.5MeV 20MeV	1MeV 20MeV
1.217	6.1	8.45E+17	1.01E+18	5.67E+18	4.18E+18	1.69E+19	9.38E+18	4.84E+18
1.119	4.8	2.13E+18	2.56E+18	1.29E+19	9.62E+18	3.98E+19	2.19E+19	1.12E+19
0.233	5	6.16E+18	7.40E+18	3.72E+19	2.78E+19	1.15E+20	6.35E+19	3.28E+19
0.233	3.3	9.68E+19	1.16E+20	1.42E+20	3.88E+19	1.20E+20	6.55E+19	3.35E+19
0.107	2.7	1.24E+18	1.49E+18	1.82E+18	1.99E+19	1.21E+20	6.57E+19	3.29E+19

Table 4. Fast neutron fluence ($1/cm^2$) in the container axis at various elevation above core midplane, chain No.2.

Elevat. [m]	SD %	0 0.1eV	0 0.398eV	0.398eV 10keV	10keV 0.1MeV	0.1MeV 20MeV	0.5MeV 20MeV	1MeV 20MeV
1.23	1.9	5.85E+17	7.02E+17	8.59E+17	1.00E+18	5.11E+18	2.78E+18	1.40E+18
1.215	5.9	2.17E+18	2.61E+18	3.19E+18	2.90E+18	1.43E+19	7.88E+18	4.06E+18
1.116	3.1	5.67E+18	6.80E+18	8.32E+18	8.41E+18	4.20E+19	2.30E+19	1.17E+19
0.99	4.1	4.82E+18	5.79E+18	7.08E+18	1.27E+19	6.85E+19	3.73E+19	1.88E+19
0.609	5.4	1.15E+19	1.38E+19	1.69E+19	2.08E+19	1.08E+20	5.88E+19	2.99E+19
0.102	2.9	1.24E+19	1.49E+19	1.83E+19	2.35E+19	1.23E+20	6.72E+19	3.44E+19
-0.41	5.5	1.33E+19	1.59E+19	1.95E+19	2.38E+19	1.23E+20	6.74E+19	3.44E+19
-0.79	4.9	1.08E+19	1.30E+19	1.59E+19	2.06E+19	1.07E+20	5.87E+19	2.99E+19
-0.91	2.4	9.52E+18	1.14E+19	1.40E+19	1.89E+19	9.82E+19	5.34E+19	2.68E+19
-1.04	4.7	7.80E+18	9.37E+18	1.15E+19	1.63E+19	8.58E+19	4.69E+19	2.38E+19

Paper 5: "FAST NEUTRON FLUENCE MONITORING ON NPP DUKOVANY"

As stated by Hogel et al. in their Prague paper [Ho96a]:

"The results and experiences gained by the ex-vessel fast neutron fluence measurements performed at the nuclear power plant (NPP) Dukovany are summarized in the present paper. The continuous monitoring is obligatory for all four units of NPP. Insertion of the activation detectors into the reactor cavity and their withdrawal are performed during the refueling outage. The time corrections for the decay during the irradiation were performed with respect to the time course of reactor power . The neutron spectra measured by differential methods on mock-up experiments in the LR-O experimental reactor were used to determine the fast neutron flux density and fluence in the reactor cavity, on the inner surface, and 1/4 T of the RPV thickness. Discussion of results and uncertainty propagation is performed.

***INTRODUCTION** -- The knowledge of fast neutron spectrum and fluence impinging on the reactor pressure vessel (RPV) of nuclear power reactors is one of the basic criteria for evaluation of its lifetime. The monitoring of the neutron fluence on the RPV inner surface in VVER 440 reactors can be performed only indirectly, either in the surveillance containers, or on the RPV outer surface, in the reactor cavity. In both cases we have to know the conversion factors for determination of neutron fluence on the inner surface and on the 1/4 of the pressure vessel thickness, respectively. These factors can he obtained experimentally in mock-up experiments or from calculations*

The standard surveillance program on all four units of NPP Dukovany was finished after five cycles. But it was not too suitable for determination of the neutron fluence on the RPV inner surface. In essence, it was not even designed for this purpose. First of all, the detector choice was also considerably limited. Secondly, the detectors were placed eccentrically in the surveillance containers which can cause, together with unknown orientation of the container in relation to the core, a systematic error up to 20% in neutron fluence in surveillance specimen positions.

Ex-vessel fluence monitoring in the reactor cavity has many indisputable advantages. Detectors can be located at any place behind the RPV and therefore it is possible to obtain the complete azimuthal and vertical neutron fluence distribution. They are replaced after each cycle which also enables to use nuclear reactions having a shorter life-time (nickel, titanium, f.p.), and activation detectors can be readily replaced or added into the detector set. The measurement and the evaluation can also be performed in a relatively short time period after the end of cycle (~ one month). The fact that the conversion factors for critical points of the RPV can be obtained more accurately and reliably from this measurements then from the detectors located in the surveillance containers is considered to be a big advantage as well. ***Regular fast neutron fluence monitoring in the reactor cavity started in 1992 and one can just pity that it was not performed since the beginning of the operation as a logical supplement to the surveillance program.***

***FLUENCE DETERMINATION IN RPV CRITICAL POINTS** -- There are two critical points of VVER 440 RPV. The azimuthal and vertical maximum of neutron fluence, and the weld No4 lying in the lower part of the reactor core. The spatial neutron fluence distribution measurements enable us to find the location of the fluence maximum and also to determine the conversion* factor for the weld position.

Two methods are used to obtain the neutron fluences from the measured reaction rates of the activation detectors. In both of them, the spectra measured by Holman [Ho91], Jansky and Marek [Bu86] on the VV... 440 mock-up using the differential proton recoil method, play the decisive role. These experiments were carried out on the LR-O Benchmark Reactor in NRI-Rez.

In the first method, the so-called effective cross sections σ_{ef} are calculated for single detectors using the definition...So we can write

$$\Phi_{ef} = Reaction\ Rate / \sigma_{ef}$$

where Φ_{ef} is the neutron flux density.

Using the evaluated reaction rates and the effective cross sections calculated in spectra measured on the mock-up, we can simply obtain the neutron flux density above the chosen energy for all used activation monitors. It is obvious that this method can be used only when the neutron spectrum is well known.

The second way of experimental data handling is the spectrum unfolding from evaluated reaction rates using the SAND II code, where the spectra measured on mock-ups by differential methods are used as the guess spectra. The neutron flux densities and fluences are calculated from the resulting spectra. In all cases the dosimetry sensor differential cross sections from IRDF 90 library are used for the evaluation.

The values of fast neutron flux densities over 1 MeV in the reactor cavity from two different measurements gained by both methods are presented in Table 2. Good agreement of results obtained from nuclear reactions having considerable different energy responses confirms the applicability of spectra measured on the mock-up for the activation data evaluation.

Finally, the fluences on the inner surface of the pressure vessel and in 1/4 of RPV thickness are obtained by multiplying the values evaluated for the outer surface by attenuation factors obtained from the mock-up experiment. The factors were measured using both differential and activation methods.

DISCUSSION OF RESULTS -- The evaluated maximal fluences above 1 MeV on the RPV inner surface in measured cycles are in the range $3.96\text{-}4.82 \times 10^{22}$ m^{-2}, depending on the power distribution in the core and the cycle length.

The following main components contribute to the total uncertainty of the evaluated fast neutron fluence on the inner surface of RPV:

- The uncertainty of the measured activities and evaluated reaction rates,
- the evaluation of the neutron flux density and fluence from reaction rates, and
- the uncertainty of the attenuation factor.

As to the first item, we can say that the determination of the uncertainty of reaction rates is *satisfactorily solved. In Table 3 there are the variances and the covariance matrix of the evaluated reaction rates from the measurements performed on Unit 1. Cycle 8. The list of partial components is presented in Table 4. Some of these values (time course of reactor power, recalculation to the maximum) are fully correlated. The correlation coefficients of gamma detector efficiencies are given by the program ETA, and the remaining values are supposed to be uncorrelated.*

The missing covariance matrices of differential cross sections and ones of the spectra measured on the mock-ups doesn't allow us to determine statistically correctly the remaining two components.

In Table 5 there are the ratios of fluences obtained from single detectors using the effective cross section method to fluences obtained from SAND II. The rather small dispersion of the presented values in the upper part of table confirms the applicability of spectra measured on the mock-up for our evaluation of the activation data. Nb, Np and U detectors were not included in the SAND II unfolding, as we have performed only a few measurements and we are not sure the measuring and evaluation process is fully correct.

The uncertainty of the attenuation factor given by Holman is 15% for the threshold 0.5 MeV and 10% for 1.0 MeV. Our conservative estimate of the total uncertainty of the fast neutron fluence on the inner surface of the RPV is 15% for 0.5 MeV and 10% for 1.0 MeV."

Tab 2: Measured reaction rates and evaluated fast neutron flux densities. The comparison of results obtained using effective cross section method and from SAND II code

Reaction	Unit 1, cycle 9		Unit 4, cycle 7	
	rr [s^{-1}]	$\Phi_{1,0}$ [m^{-2} s^{-1}]	rr [s^{-1}]	$\Phi_{1,0}$ [m^{-2} s^{-1}]
^{54}Fe(n,p)	1.18E-15	2.07E+14	1.03E-15	1.81E+14
^{58}Ni(n,p)	1.58E-15	1.98E+14	1.42E-15	1.77E+14
^{46}Ti(n,p)	1.96E-16	2.23E+14	1.68E-16	1.92E+14
^{63}Cu(n,α)	1.23E-17	2.15E+14	1.07E-17	1.87E+14
^{55}Mn(n,2n)	1.28E-17	1.82E+14	1.19E-17	1.69E+14
^{89}Y(n,2n)	1.07E-17	1.95E+14		
Mean value		2.03E+14		1.82E+14
SAND II		2.04E+14		1.79E+14

Tab 3. Reaction rates, their variances and correlation matrix from the measurement in the unit 1, cycle 8.

Reaction	rr [s^{-1}]	δ [%]	Correlation matrix (x100)			
^{54}Fe(n,p)	1.163E-15	3.2	100			
^{58}Ni(n,p)	1.599E-15	2.8	62	100		
^{46}Ti(n,p)	1.856E-16	2.6	67	78	100	
^{63}Cu(n,a)	1.184E-17	2.6	67	78	84	100

Tab 4 Partial components for calculation of variances and correlation matrix of evaluated reaction rates.

Reaction	Variance [%]							
	eta	eps	g	N	K	M	N_0	ssg
$^{54}Fe(n,p)^{54}Mn$	0.77 [b]	-	0.1	1	1 [a]	2 [a]	1.7	-
$^{58}Ni(n,p)^{58}Co$	0.77 [b]	0.3	0.1	1	1 [a]	2 [a]	-	-
$^{46}Ti(n,p)^{46}Sc$	0.76 [b]	-	-	1	1 [a]	2 [a]	-	0.2
$^{63}Cu(n,a)^{60}Co$	0.77 [b]	-	-	1	1 [a]	2 [a]	-	0.2

eta detection efficiency
eps gamma branching ratio
g foil mass
N counting statistics
K time course of irradiation
M recalculation to the maximum
N_0 number of target nuclei in the foil
ssg gamma self-absorption in the foil
a fully correlated
b corr(eta1,eta2)= corr(eta1,eta3)= 1.00
corr(eta2,eta3)= corr(eta3,eta4)= 1.00
corr(eta1,eta4)= 0.94
corr(eta2,eta4)= 0.93

Tab.5. The ratio of fluences obtained from single monitors to fluences evaluated by SANDII (without Nb, Np and U)

Unit	1			2			3			4		
Cycle	8	9	10	8	9	10	7	8	9	6	7	8
Fe	1.01	1.04	0.93	0.94	0.98	0.96	0.98	0.96	1.02	1.04	0.98	0.97
Ni	0.97	1.00	0.95	0.98	0.97	1.06	1.01	0.99	1.00	1.02	0.95	0.94
Ti	1.02	1.06	1.05	1.06	1.05	1.07	1.06	1.04	1.06	1.06	1.03	1.02
Cu	1.00	1.07	0.96	1.00	1.02	1.03	1.01	0.99	1.04	1.00	1.02	1.01
Mn		0.86	0.88	0.90	0.96	0.92	0.86	0.85	0.91	0.88	0.94	0.93
Y		0.93				0.97						
Np		0.88							0.93			
U		0.90							1.04			
Nb		1.27	1.23	1.12	1.03		1.08	1.18			1.08	1.14

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ENCLOSURE 5

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

Palisades Reactor Vessel Fluence

Resume & Independent Review
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Curiculum Vitae

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   Proceedings of the 7[th] ASTM-EURATOM Symposium on Reactor Dosimetry 1992

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