NUREG/CR-2805, Vol. 3
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LWR PRESSURE-VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

1982 ANNUAL REPORT
(OCTOBER 1, 1981 - SEPTEMBER 30, 1982)

Hanford Engineering Development Laboratory

Prepared by
W.N. McElroy (HEDL)
F.B.K. Kam (ORNL)
J.A. Grundl and E.D. McGarry (NBS)
A. Fabry (CEN/SCK)

Prepared for the U.S. Nuclear Regulatory Commission

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Operated by Westinghouse Hanford Company
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A Subsidiary of Westinghouse Electric Corporation

Prepared by
W.N. McElroy (HEDL)
F.B.K. Kam (ORNL)
J.A. Grundl and E.D. McGarry (NBS)
A. Fabry (CEN/SCK)

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FOREWORD

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) has been established by NRC to improve, test, verify, and standardize the physics-dosimetry-metallurgy, damage correlation, and the associated reactor analysis methods, procedures and data used to predict the integrated effect of neutron exposure to LWR pressure vessels and their support structures. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the US NRC-supported activities at HEDL, ORNL, NBS, and MEA-ENSA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Jülich, Germany), and several UK laboratories have been extended to a number of other countries and laboratories. These cooperative links are strengthened by the active membership of the scientific staff from many participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR surveillance standards.

The primary objective of this multilaboratory program is to prepare an updated and improved set of physics-dosimetry-metallurgy, damage correlation, and associated reactor analysis ASTM Standards for LWR pressure vessel and support structure irradiation surveillance programs. Supporting this objective are a series of analytical and experimental validation and calibration studies in "Standard, Reference, and Controlled Environment Benchmark Fields," research reactor "Test Regions," and operating power reactor "Surveillance Positions."

These studies will establish and certify the precision and accuracy of the measurement and predictive methods recommended in the ASTM Standards and used for the assessment and control of the present and end-of-life (EOL) condition of pressure vessel and support structure steels. Consistent and accurate measurement and data analysis techniques and methods, therefore, will be developed, tested and verified along with guidelines for required neutron field calculations used to correlate changes in material properties with the characteristics of the neutron radiation field. It is expected that the application of the established ASTM Standards will 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.

The assessment of the radiation-induced degradation of material properties in a power reactor requires accurate definition of the neutron field from the outer region of the reactor core to the outer boundaries of the pressure vessel. Problems with measuring neutron flux and spectrum are associated with two distinct components of LWR irradiation surveillance procedures: 1) proper application of calculational estimates of the neutron exposure at in- and ex-vessel surveillance positions, various locations in the vessel wall and ex-vessel support structures, and 2) understanding the relationship between material property changes in reactor vessels and their support structures, and in metallurgical test specimens irradiated in test reactors and at accelerated neutron flux positions in operating power reactors.

The first component requires verification and calibration experiments in a variety of neutron irradiation test facilities including LWR-PV mockups, power reactor surveillance positions, and related benchmark neutron fields. The benchmarks serve as a permanent reference measurement for neutron flux and fluence detection techniques, which are continually under development and widely applied by laboratories with different levels of capability. The second component requires a serious extrapolation of an observed neutron-induced mechanical property change from research reactor "Test Regions" and operating power reactor "Surveillance Positions" to locations inside the body of the pressure vessel wall and to ex-vessel support structures. The neutron flux at the vessel inner wall is up to one order of magnitude lower than at surveillance specimen positions and up to two orders of magnitude lower than for test reactor positions. At the vessel outer wall, the neutron flux is one order of magnitude or more lower than at the vessel inner wall. Further, the neutron spectrum at, within, and leaving the vessel is substantially different.

In order to meet the reactor pressure vessel radiation monitoring requirements, a variety of neutron flux and fluence detectors are employed, most of which are passive. Each detector must be validated for application to the higher flux and harder neutron spectrum of the research reactor "Test Region" and to the lower flux and degraded neutron spectrum at "Surveillance Positions." Required detectors must respond to neutrons of various energies so that multigroup spectra can be determined with accuracy sufficient for adequate damage response estimates. Detectors being used, developed and tested for the program include radiometric (RM) sensors, helium accumulation fluence monitor (HAFM) sensors, solid state track recorder (SSTR) sensors, and damage monitor (DM) sensors.

The necessity for pressure vessel mockup facilities for physics-dosimetry investigations and for irradiation of metallurgical specimens was recognized early in the formation of the NRC program. Experimental studies associated with high and low flux versions of a PWR pressure vessel mockup are in progress in the US, Belgium, and the United Kingdom. The US low flux version is known as the ORNL Poolside Critical Assembly (PCA) and the high flux version is known as the ORR Poolside Facility (PSF). Both are located at Oak Ridge, Tennessee. As specialized benchmarks, these facilities are providing well-characterized neutron environments where active and passive neutron dosimetry, various types of LWR-PV and support structure neutron field calculations, and temperature-controlled metallurgical specimen exposures are brought together. The two key low flux pressure vessel mockups in Europe are known as the Mol-Belgium-VENUS and Winfrith-United Kingdom-NESDIP facilities. The VENUS facility is to be used for PWR core source and azimuthal lead factor studies while NESDIP is to be used for PWR cavity and azimuthal lead factor studies.

The results of the measurement and calculational strategies outlined here will be made available for use by the nuclear industry as ASTM Standards. Federal Regulation 10CFR50 already requires adherence to several ASTM Standards that establish a surveillance program for each power reactor and incorporate metallurgical specimens, physics-dosimetry flux-fluence monitors and neutron field evaluation. Revised and new standards in preparation will be carefully up-dated, flexible, and, above all, consistent.

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Prepared by W. N. McElroy, F. B. K. Kam, J. A. Grundl, E. D. McGarry, and A. Fabry with reference to and/or use of selected contributions from the following participants and laboratories:

W. N. McElroy, L. D. Blackburn, R. Gold, G. L. Guthrie,
B. J. Kaiser, L. S. Kellogg, E. P. Lippincott, W. Y. Matsumoto,
J. P. McNeese, C. C. Preston, J. H. Roberts, F. H. Ruddy,
J. M. Ruggles, F. A. Schmittroth, R. L. Simons, and J. A. Ulseth
Hanford Engineering Development Laboratory (HEDL), USA

J. A. Grundl, C. M. Eisenhauer, G. P. Lamaze, and E. D. McGarry
National Bureau of Standards (NBS), USA

F. B. K. Kam, R. E. Maerker, and F. W. Stallmann
Oak Ridge National Laboratory (ORNL), USA

A. Fabry, J. Debrue, G. DeLeeuw, S. DeLeeuw, G. Minsart, and H. Tourwé
Centre d'Etudes de l'Energie Nucleaire,
Studiecentrum voor Kernenergie (CEN/SCK), Mol, Belgium

W. Schneider, L. Weise, and D. Packur
Kernforschungsanlage (KFA) Jülich, Federal Republic of Germany

M. Austin, A. F. Thomas, and T. J. Williams
Rolls-Royce & Associates Limited (RRAL), UK

A. J. Fudge (Harwell) and J. Butler (Winfrith)
Atomic Energy Research Establishment (AERE), UK

A. A. Alberman and J. P. Genthon (Saclay) and P. Mas (Grenoble)
Commissariat a l'Energie Atomique,
Centre d'Etudes Nucleaires (CEA/CEN), France

H. Farrar IV and B. L. Oliver
Rockwell International (RI), USA

W. Zimmer
EG&G ORTEC, USA

G. R. Odette and G. Lucas
University of California at Santa Barbara (UCSB), USA

P. D. Hedgecock
RDH Engineering, USA

J. S. Perrin and R. A. Wullaert
Fracture Control Corporation (FCC), USA

B. A. Magurno and J. Carew
Brookhaven National Laboratory (BNL), USA

T. U. Marston, O. Ozer, T. Passell, and R. Shaw
Electric Power Research Institute (EPRI), USA

C. Cogburn and L. West
University of Arkansas (UA), USA

W. C. Hopkins
Bechtel Power Corporation, USA

J. R. Hawthorne
Materials Engineering Associates (MEA), USA

E. B. Norris
Southwest Research Institute (SWRI), USA
Battelle Memorial Institute (BMI) Columbus Laboratory, USA

S. L. Anderson, T. R. Mager, and S. E. Yanichko
Westinghouse Electric Corporation (W), USA

A. A. Lowe, Jr., C. L. Whitmarsh, and R. H. Lewis
Babcock & Wilcox Company (B&W), USA

G. C. Martin, Jr.
General Electric Company (GE), USA

G. P. Cavanaugh, J. D. Varsik, S. T. Byrnes, and J. J. Koziol
Combustion Engineering, Inc. (CE), USA

N. Tsoulfanidis and D. R. Edwards
University of Missouri, Rolla (UMR), USA

S. Grant
Carolina Power and Light Company, USA

H. F. Jones
Maine Yankee Atomic Power Company, USA

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F. B. Kam (ORNL)
A. Fabry (CEN/SCK)

ABSTRACT

This report describes progress made in the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) during FY 1982. The primary concern of this program is to improve, test, verify and standardize the physics-dosimetry-metallurgy and the associated reactor and damage analysis procedures and data used for predicting the integrated effects of neutron exposure to LWR pressure vessels and support structures. These procedures and data are being recommended in a new and updated set of ASTM standards being prepared, tested, and verified by program participants. These standards together with parts of the US Code of Federal Regulations and ASME codes are needed and used for the assessment and control of the condition of LWR pressure vessels and support structures during the 30 to 50 year lifetime of a nuclear power plant.

ACKNOWLEDGMENTS

The success of the LWR Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) continues to depend on the efforts and the free exchange of ideas and views by representatives of a large number of research, service, regulatory, vendor, architect/engineer and utility organizations. The information reported herein could not have been developed without the continuing support of the respective funding organizations and their management and technical staffs. Special acknowledgment is due to C. Z. Serpan of NRC for having identified the need for an international program such as the LWR-PV-SDIP and for making it possible by taking a strong overall support and management lead.

Additional acknowledgment is due to D. G. Doran and W. F. Sheely of HEDL for their technical reviews and constructive comments. Very special acknowledgment is given to N. E. Kenny and J. M. Dahlke, who edited this document and to the HEDL Publications Services, Word Processing, Graphics, and Duplicating personnel, who contributed to its preparation.

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ACRONYMS

AERE	Atomic Energy Research Establishment
ANO-1,-2	Arkansas Nuclear One, Units 1 and 2 PWR Nuclear Power Plants
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&W	Babcock & Wilcox
BMI	Battelle Memorial Institute
BNL	Brookhaven National Laboratory
BR-2	Belgium Test Reactor
BR-3	Belgium PWR Nuclear Power Plant
BSR	Bulk Shielding Reactor
BWR	Boiling Water Reactor
CE	Combustion Engineering
CEA/CEN	Centre d'Etudes Nucleaires (Saclay and Grenoble, France)
CEN/SCK	Centre d'Etudes de l'Energie Nucleaire (Mol, Belgium)
CR	Crystal River PWR Nuclear Power Plant
DB	Davis-Besse PWR Nuclear Power Plant
DBTT	Ductile-Brittle Transition Temperature
DIDO	UK Test Reactor
DM	Damage Monitor
DPA	Displacements Per Atom
EBR-II	Experimental Breeder Reactor II
EFPY	Effective Full-Power Years
ENDF	Evaluated Nuclear Data File
ENSA	Engineering Services Associates
EPRI	Electric Power Research Institute
FBR	Fast Breeder Reactor
FCC	Fracture Control Corporation
FRJ 1	German Test Reactor
FRJ 2	German Test Reactor
FSAR	Final Safety Analysis Review
GE	General Electric
HAFM	Helium Accumulation Fluence Monitor
HEDL	Hanford Engineering Development Laboratory

ACRONYMS (Cont'd)

HERALD	UK Test Reactor
HOTS	Hanford Optical Track Scanner
HSST	Heavy Section Steel Technology Program
IAEA	International Atomic Energy Agency
KFA	Kernforschungsanlage (Jülich, Germany)
LWR	Light Water Reactor
MATSURV	NRC Computerized Reactor Pressure Vessel Materials Information System
MEA	Materials Engineering Associates, Inc.
MFR	Magnetic Fusion Reactor
MPC	Materials Property Council, Subcommittee 6 on Nuclear Materials
NBS	National Bureau of Standards
NDTT	Nil Ductility Transition Temperature
NESDIP	PWR Mockup (Winfrith, United Kingdom)
NRC	Nuclear Regulatory Commission
OCA	Overcooling Accident
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Research Reactor at ORNL
PCA	Poolside Critical Assembly at ORNL
PCBT	PV-Simulator, "Emplacement Special" Position in the Melusine Reactor (Grenoble, France)
PSF	Poolside Facility at ORNL
PV	Pressure Vessel
PVF	Pressure Vessel Front
PVS	Pressure Vessel Simulator (Includes both SPVC and SVBC)
PWR	Pressurized Water Reactor
QA	Quality Assurance
RI	Rockwell International
RIL	Research Information Letter
RM	Radiometric Monitors
RRAL	Rolls-Royce & Associates Limited (UK)

ACRONYMS (Cont'd)

RT_{NDT}	Reference Nil Ductility Transition Temperature
RT_{NDT_o}	Initial Unirradiated Value of RT_{NDT}
RPV	Reactor Pressure Vessel
SDIP	Surveillance Dosimetry Improvement Program
SDMF	Simulated Dosimetry Measurement Facility
SEM	Scanning Electron Microscope
SPVC	Simulated Pressure Vessel Capsule
SRM	Standard Reference Material
SS	Stainless Steel
SSC	Simulated Surveillance Capsule
SSTR	Solid State Track Recorder
SUNY-NSTF	State University of New York-Nuclear Science Technology Facilities at Buffalo, NY
SVBC	Simulated Void Box Capsule
TM	Temperature Monitor
TSB	Thermal Shield Back
UCSB	University of California at Santa Barbara
UA	Univeristy of Arkansas
UK	United Kingdom
UMR	University of Missouri, Rolla
VENUS	PWR Mockup at Mol, Belgium
WEC (or W)	Westinghouse Electric Corporation
WRSR	Water Reactor Safety Research

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1.0 INTRODUCTION

Light water reactor pressure vessels (LWR-PV) are accumulating significant neutron fluence exposures, with consequent changes in their steel fracture toughness and embrittlement characteristics. Recognizing that accurate and validated measurement and data analysis procedures are needed to periodically evaluate the metallurgical condition of these reactor vessels, the US Nuclear Regulatory Commission (NRC) has established the LWR Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP). The primary concerns of this program are to improve, test, verify, and standardize 1) the physics-dosimetry-metallurgy, 2) the damage correlation, and 3) the associated reactor analysis methods, procedures and data used for predicting the integrated effects of neutron exposure to LWR pressure vessels and support structures.

A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the US NRC supported activities at HEDL, ORNL, NBS and MEA-ENSA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Jülich, Germany) and several UK laboratories have been extended to a number of other countries and laboratories. (A current listing to the literature of documents most relevant to LWR-PV-SDIP interlaboratory efforts up to October 1982 is provided in References 1-93.) These cooperative links have been strengthened by the active membership of the scientific staff of many of the participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications.[94] Several subcommittees of ASTM E10 are responsible for the preparation of LWR pressure vessel and support structure surveillance standards. Summary information on LWR-PV-SDIP FY 1982 research results are provided in Section 2.0.

As discussed in Sections 3.1, 3.2, and 3.3, the <u>major benefit</u> of this program will be a significant improvement in the accuracy of the assessment and control of the present and end-of-life (EOL) condition of light water reactor pressure vessels and their support structures. A <u>primary objective</u> of this multilaboratory program is to prepare an updated and improved set of physics-dosimetry-metallurgy, damage correlation, and the associated reactor analysis ASTM standards for LWR pressure vessel and support structure surveillance programs, as described in Section 3.4.1. <u>Supporting this objective</u> are a series of analytical and experimental verification and calibration studies in "Benchmark Neutron Fields," research reactor "Test Regions," and operating power reactor "Surveillance Positions." As discussed in Sections 3.4.2, 3.4.3, and 3.5, these studies will establish and certify the precision and accuracy of the measurement and predictive methods recommended for use in the ASTM standards. Consistent and accurate measurement and data analysis techniques and methods, therefore, will have been developed, tested, and verified along with guidelines for required neutron field physics-dosimetry-metallurgy calculations. Based on nuclear power plant operational, safety, licensing, and regulatory requirements, these calculations are then used 1) to correlate changes in material properties with the characteristics of the neutron radiation field and 2) to predict the present and EOL condition of pressure vessel and support structure steels from both power and research reactor data.

2.0 FY 1982 RESEARCH RESULTS - SUMMARY

To account for neutron radiation damage in setting pressure-temperature limits and making fracture analysis[1-8,13-22,24,39,46,57,60-65,67-70,83-86,88,91-93] neutron-induced changes in reactor pressure vessel steel fracture toughness and embrittlement must be predicted, then checked by extrapolation of surveillance program data during the vessel's service life. Uncertainties in the predicting methodology can be significant. The main variables of concern are associated with:

- Steel chemical composition and microstructure
- Steel irradiation temperature
- Power plant configurations and dimensions - core edge to surveillance to vessel wall to support structure positions
- Core power distribution
- Reactor operating history
- Reactor physics computations
- Selection of neutron exposure units
- Dosimetry measurements
- Neutron spectral effects
- Neutron dose rate effects.

Variables associated with the physical measurements of PV steel property changes are not considered here and are addressed separately in Appendices G and H of 10 CFR Part 50,[13] in ASTM Standards,[94] and elsewhere[2-9,14-19,62,64,65,69,70,85,86,91,92].

The US NRC had previously estimated that there were approximately 21 operating early generation US pressurized water reactors (PWR) that might have beltline materials with marginal toughness, relative to the existing requirements of Appendices G and H and Regulatory Guide 1.99,[8] sometime within their service life;[90] i.e., in the range up to about 32 years.

As older vessels become more highly irradiated, the predictive capability for changes in fracture toughness and embrittlement must improve, particularly for plants operated beyond their current design service life; i.e., in the range above about 32 years. Since during the vessel's service life an increasing amount of information will be available from research reactor test and power reactor surveillance programs, better procedures to evaluate and use this information can and must be developed. The most appropriate way to make information available on these procedures is through voluntary consensus standards, such as those now being developed by ASTM Committee E10 on Nuclear Technology and Applications[17,94] and discussed here and in Sections 2.1 and 3.4.

Important summary highlights of FY 1982 research activities of this multi-laboratory program are:

A. The completion of first, revised, or final drafts (Figures 3.10 and 3.11) of fifteen (✓) of twenty-one ASTM standards which focus on the physics-dosimetry-metallurgy, damage correlation, and the associated reactor analysis and interpretation aspects of the problem of guaranteeing the safety and integrity of the pressure vessel boundary and its support structures for LWR power reactors,[17] see Section 2.1.1.

B. Initiation and completion of important supporting verification and calibration benchmark studies, reviews, and neutron and gamma field experimental and calculational work, Tables 3.5 and 3.9, which demonstrate and verify the direct applicability of the twenty ASTM standards (nine "practices", six "guides", and five "methods").[10,11,18,21-22,24-29,31,33-38,40-43,47-54,56,59-70,72-76,80-89,94] See Sections 2.0, 3.0 and 5.0.

Of particular interest here was 1) the completion of studies on fuel management effects and neutron exposure parameters and their impact for PV pressurized thermal shock studies related to the assessment and control of the present and EOL condition of pressure vessel and support structure steels, as discussed in References 10, 11, 22, and 63, and Sections 3.3 and 5.0 and 2) the initiation and planning of verification tests in H. B. Robinson, Maine Yankee, and Crystal River (or Davis-Besse), see Table 3.9.

C. The completion of key experimental physics-dosimetry studies associated with the ORNL PCA low flux version of a PWR pressure vessel mock-up[24-26,50,51,76,80] and the start of work associated with the VENUS[18,24,60] and NESDIP mockups[24,29,61] (Figures 3.43 and 3.44 and Sections 2.2 and 3.4.3.3), in Belgium and the United Kingdom, respectively.

D. The successful completion of the 2 years of irradiations and initial testing and analyses for the Oak Ridge Research Reactor simulated surveillance capsule (SSC), simulated pressure vessel capsule (SPVC) and simulated void box capsule (SVBC) LWR power plant physics-dosimetry-metallurgy mockup experiments.[31,41-43,47,69,71]

E. The completion of required studies associated with the evaluation and reevaluation of exposure units and values for existing and new metallurgical data bases (NRC, MPC, EPRI, ASTM and others),[6-9,62,91,92] Figures 3.13 through 3.42 and Tables 3.1 through 3.13. The initial power reactor studies have involved the reanalysis of data from 41 PWR surveillance capsule reports for Westinghouse, Babcock and Wilcox, and Combustion Engineering power plants. Using a consistent set of auxiliary data and dosimetry-adjusted reactor physics results, the revised fluence values for $E > 1$ MeV averaged 29% higher than the originally reported values. The range of fluence values (new/old) was from a low of 0.82 to a high of 2.44, see Table 3.4 of Section 3.0 and Reference 40. The initial research reactor studies have involved the reanalysis of data originally reported by NRL and HEDL, see Section 2.4.2 and the Appendix, Section 5.0, and its references.

F. The completion of required studies associated with the data development and testing for new trend curves for the ΔRT_{NDT} shift versus neutron exposure (fluence E > 1.0 MeV and dpa) for an NRC selected power reactor surveillance capsule data base of 138 points, see Section 2.4.1 and References 6, 7, 8, 74, 75, and 91. The status of EPRI supported program work related to physics-dosimetry-metallurgy data development and testing is provided in References 2, 62, 64, and 65.

G. The completion of the planning, work, preparation of papers, presentations, and documentation of the proceedings for the Fourth ASTM-EURATOM International Symposium on Reactor Dosimetry held at NBS in March, 1982; see Reference 31; also US, Belgium, and UK papers for the NRC 10th WRSR Information Meeting held at NBS in October 1982.

2.1 ASTM STANDARDS AND PROGRAM DOCUMENTATION

2.1.1 ASTM Standards

Figures 3.10 and 3.11 of Section 3.0 provide information on the interrelationships and current schedule for the preparation and acceptance of the set of 21 ASTM standards. Results of ASTM balloting for these standards were discussed at the January 1982 Houston and June 1982 Scottsdale, ASTM E10 Meetings. Figures 3.10 and 3.11 will be updated next at the January 1983 Orlando meeting and will be reviewed by the ASTM E10.05 Nuclear Radiation Metrology and E10.02 Metallurgy Subcommittee members to coordinate the preparation, balloting, testing, and acceptance of the entire set of standards. Reference 17 provides additional information related to the scope, content, and preparation of most of these standards. More detailed, but summary information on the status of the preparation of the individual standards follows:

E706(0) Master Matrix Guide

Lead Authors	W. McElroy (E10.05)* and P. Hedgecock (E10.02)*
Participants	Lead authors of all Practices (I), Guides (II), and Methods (III)
Status	In place in 1982 Annual Book of Standards as E706-81a. Scope and discussion sections must be reviewed and updated as necessary by lead authors of I, II, and III for Orlando meeting. Lead authors are also to update Figure 3.10 (Matrix) and Figure 3.11 (Schedule) as well as lists of applicable documents for each standard.

E706(IA) Analysis and Interpretation of Reactor Surveillance Results

Lead Authors	S. Anderson and W. McElroy (E10.05)
Participants	E. Lippincott, G. Guthrie, F. Schmittroth, P. Hedgecock, O. Ozer, C. Whitmarsh, G. Cavanaugh, G. Martin, E. Norris C. Serpan, R. Gold, L. Kellogg, F. Ruddy, J. Roberts, B. Oliver, H. Farrar, J. Perrin, M. Austin, A. Thomas, A. Fabry, H. Tourwé, and A. Fudge
Status	Received ASTM Society approval. Now designated as E853-81 and appears in 1982 Annual Book of Standards. Needs updating of list of applicable documents and mention of additional new standards as appropriate, see Table 3.2 of Section 3.0.

*P. D. Hedgecock and W. N. McElroy are the current chairmen of the E10.02 and E10.05 Subcommittees, respectively, of the ASTM E10 committee. The current chairman of the ASTM E10 Committee is R. H. Lewis.

E706(IB) Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials

Lead Authors	J. Beeston (E10.02); E. Norris and H. Farrar (E10.05)
Participants	W. McElroy, P. Hedgecock, E. Lippincott, B. Oliver, C. Serpan, and E10.02 members
Status	E184-79 is on the books. E. Norris and H. Farrar are expected to update the physics-dosimetry parts of the standard for the Orlando meeting. E10.02 and E10.05 members should actively pursue the revision because this standard provides the interface between all "metallurgy" and "physics-dosimetry" standards and applies to LWRs, FBRs, and MFRs. A title change for the standard is needed such as to "Recommended Physics-Dosimetry-Metallurgy Interface Standard for LWR, FBR, MFR Development Programs."

E706(IC) Surveillance Test Results Extrapolation

Lead Authors	G. Guthrie and S. Anderson (E10.05); S. Byrne (E10.02)
Participants	F. Stallmann, O. Ozer, R. Maerker, P. Hedgecock, W. McElroy, E. Lippincott, C. Serpan, N. Randall, C. Whitmarsh, G. Cavanaugh, W. Hopkins, M. Austin, N. Tsoulfanidis, L. Kellogg, R. Gold, F. Ruddy, J. Roberts, E. McGarry, J. Wagschal, G. Martin, A. Fabry, and C. Eisenhauer
Status	A draft has been prepared and will be balloted in CY 1983 by E10.02 and E10.05, incorporating comments received at Scottsdale. This practice has been given the number E560 by ASTM, which is the number of the present standard (E560-77) it will replace. Information on physics-dosimetry-metallurgy studies from test and power reactor benchmark studies supporting the preparation of this standard are provided in subsequent sections of this annual report.

E706(ID) Displaced Atom (DPA) Exposure Unit

Lead Authors	D. Doran and E. Lippincott (E10.05)
Participants	W. McElroy, G. Guthrie, F. Schmittroth, A. Thomas, R. Dierckx, O. Ozer and W. Zijp
Status	Accepted as standard and appears in 1982 Annual Book of Standards as E693-79. The need exists to update the basic nuclear data, i.e., using ENDF/B-V data and comparing the results with those obtained using ENDF/B-IV data. More complete and detailed information on the testing and application of the dpa exposure unit is provided in Sections 3.3, 3.4, and 3.5. A copy of a Research Information Letter (RIL) on "An Improved Damage Exposure Unit, dpa, for LWR Pressure Vessel and Support Structure Surveillance" is provided in the Appendix, Section 5.0.

E706(IE) Damage Correlation for Reactor Vessel Surveillance

Lead Authors	G. Guthrie (E10.05) and P. Hedgecock (E10.02)
Participants	F. Stallmann, D. Doran, R. Gold, W. McElroy, S. Anderson, C. Whitmarsh, G. Cavanaugh, G. Martin, W. Hopkins, E. Norris, J. Perrin, S. Byrne, C. Serpan, N. Randall, A. Lowe, M. Austin, A. Thomas, A. Fudge, A. Fabry, and W. Schneider
Status	A draft has been prepared and requires further revision, which is dependent on the analysis of physics-dosimetry-metallurgy results from test and power reactor benchmarking studies in progress and discussed in subsequent sections of this annual report.

E706(IF) Surveillance Tests for Nuclear Reactor Vessels

Lead Authors	J. Koziol (E10.02) and C. Whitmarsh (E10.05)
Participants	E10.02 Chairman and Members for metallurgy; for physics-dosimetry update: E. Lippincott, G. Guthrie, W. McElroy, C. Serpan, E. McGarry, G. Martin, E. Norris, J. Perrin, S. Anderson, and G. Cavanaugh
Status	Appears in 1982 Annual Book of Standards as E185-82. Update on physics-dosimetry is needed in 1983. The reader is referred to Table 3.2 of Section 3.0 for information on needed changes in this key ASTM standard, which is used for establishing a physics-dosimetry-metallurgy surveillance program for each operating LWR nuclear power plant; see Reference 94.

E706(IG) Surveillance Tests for Nuclear Reactor Support Structures

Lead Authors	P. Hedgecock (E10.02) and W. Hopkins (E10.05)
Participants	E. Lippincott, G. Guthrie, W. McElroy, M. Austin, A. Thomas, S. Anderson, C. Whitmarsh, G. Cavanaugh, R. Maerker, O. Ozer, J. Wagschal, C. Serpan, N. Randall, R. Gold, F. Ruddy, J. Roberts, L. Kellogg, E. McGarry, C. Cogburn, H. Farrar, B. Oliver, J. Williams, A. Fabry and N. Tsoulfanidis
Status	A draft of the physics-dosimetry parts is to be distributed for discussion at the Orlando Meeting. A decision on metallurgy needs has yet to be made by the E10.02 Subcommittee; see Reference 93.

E706(IH) Supplemental Test Methods for Reactor Vessel Surveillance

Lead Authors	R. Hawthorne (E10.02) and E. Norris (E10.05)
Participants	E10.02 Chairman and Members; G. Guthrie, F. Stallmann and C. Serpan
Status	Will appear on December 1982 ASTM Society ballot as E636-82; see Reference 94.

E706(II) Analysis and Interpretation of Physics Dosimetry Results for Test Reactors

Lead Authors	F. Kam, F. Stallmann, and M. Williams (E10.05)
Participants	M. Austin, W. McElroy, P. Hedgecock, R. Hawthorne, G. Guthrie, A. Fabry, H. Tourwé, A. Fudge, E. Lippincott, S. Anderson, E. McGarry, B. Oliver, W. Zijp, R. Gold, F. Ruddy, J. Roberts, L. Kellogg, E. McGarry, A. Thomas, W. Schneider, M. Nakazawa, A. Sekiguchi, S. Hegedus, and G. Martin
Status	Currently being balloted by E10.05. Summary information on NRC supported US test reactor physics-dosimetry-metallurgy program studies is provided in Sections 2.2, 2.3, and 2.4. Information on other program studies is provided in Section 3.0 and References 2, 3, 9, 18, 23-29, 31, 41-43, 45-56, 60-61, 64, 66, 69-71, 76-88.

E706(IIA) Application of Spectrum Adjustment Methods

Lead Author	F. Stallmann (E10.05)
Participants	E. Lippincott, F. Schmittroth, G. Guthrie, W. McElroy, R. Maerker, M. Austin, A. Thomas, J. Wagschal, M. Nakazawa, A. Sekiguchi, J. Williams, R. Gold, O. Ozer, R. Dierckx, W. Zijp, and G. Martin
Status	Currently being balloted simultaneously by E10 and E10.05. Summary type information on the results of the application of advanced spectrum adjustment methods is provided in subsequent sections of this annual report. More detailed information will be found in References 23, 24, 31, 40, 58, 81-86, 88.

E706(IIB) Application of ENDF/A Cross Section and Uncertainty File

Lead Authors	E. Lippincott and W. McElroy (E10.05)
Participants	B. Magurno, W. Zijp, O. Ozer, R. Maerker, J. Wagschal, R. Gold, F. Ruddy, J. Roberts, A. Sekiguchi, N. Nakazawa, J. Williams, M. Austin, A. Thomas, A. Fudge, F. Stallmann, F. Kam, F. Schmittroth, H. Farrar, and B. Oliver
Status	This standard will be ready for E10 balloting in CY 1983.

A meeting of E10.05.03 Task Group on the Cross Section and Uncertainty File was held at the Fourth ASTM-EURATOM Symposium in March 1982. The discussion at this meeting centered on clarification of the ENDF/A dosimetry file makeup, since the file is an "adjusted" file. It was made clear that the cross sections will not, in general, be put in the file adjusted by "bias factors." The adjustments will be made to minimize errors using a least squares fitting procedure with various integral data. If adjustments to any cross sections are larger than the evaluated cross section uncertainty (i.e., input to the least squares procedure) the cross-section should be either flagged as unreliable or omitted from the file. This situation is unlikely to occur for those sensor reactions used for routine flux-fluence measurements.

The least-squares approach must neglect some data correlations in order to solve typical problems in reasonable amounts of computer time. The impact of neglecting these effects is presently being investigated. To the extent that these effects can be neglected, the ENDF/A cross-section file can be straight forwardly applied to flux-fluence determination problems to give not only the best estimate of the flux and spectrum, but also a reasonable estimate of the uncertainty involved.

It is anticipated that the first version of the ENDF/A file will be issued in 1983. It is apparent that the ENDF/B format may not be the most appropriate for tabulation of all the covariance data, so it may be desirable to put the data in a more appropriate format and supply a simple processing code to read the file. This will depend on the amount of covariance data to be included.

A paper on the ENDF/A file and ASTM Standard was presented at the Fourth ASTM-EURATOM Symposium,[89] and Reference 17 provides additional information on the scope of the E706(IIB) Standard.

<u>E706(IIC) Sensor Set Design and Irradiation for Reactor Surveillance</u>

Lead Authors	G. Martin and E. Lippincott (E10.05)
Participants	W. Schneider, W. McElroy, L. Kellogg, F. Ruddy, F. Schmittroth, R. Gold, J. Roberts, E. McGarry, J. Grundl, C. Whitmarsh, E. McGarry, H. Farrar, B. Oliver, A. Thomas, A. Fudge, N. Nakazawa, S. Hegedus, H. Tourwé, A. Fabry, C. Cogburn, S. Anderson, G. Cavanaugh, and A. Sekiguchi
Status	Appears in 1982 Annual Book of Standards as E844-81.

<u>E706(IID) Application of Neutron Transport Methods for Reactor Vessel Surveillance</u>

Lead Authors	L. Miller and R. Maerker (E10.05)
Participants	F. Stallmann, J. Wagschal, C. Eisenhauer, J. Grundl, E. McGarry, S. Anderson, G. Cavanaugh, C. Whitmarsh, F. Schmittroth, A. Fabry, G. Minsart, W. Hopkins, M. Austin, A. Thomas, and N. Tsoulfanidis, W. McElroy, and G. Guthrie
Status	Appears in 1982 Annual Book of Standards as E482-82.

E706(IIE) Benchmark Testing of Reactor Vessel Dosimetry

Lead Authors	E. McGarry and G. Grundl (E10.05)
Participants	C. Eisenhauer, A. Fabry, M. Austin, A. Thomas, A. Fudge, W. McElroy, L. Kellogg, E. Lippincott, J. Roberts, F. Ruddy, R. Gold, R. Dierckx, A. Sekiguchi, N. Nakazawa, E. Norris, S. Anderson, C. Whitmarsh, G. Cavanaugh, G. Martin, C. Cogburn, F. Kam, F. Stallmann, J. Williams, J. Mason, W. Zijp, B. Oliver, and H. Farrar
Status	First draft is to be submitted at Orlando meeting. A revised draft of the NBS Compendium of Benchmark Neutron Fields for Reactor Dosimetry was completed by J. Grundl of NBS and has been distributed for review by LWR-PV-SDIP participants.

E706E(IIF) Correlation of ΔNDTT with Fluence

Lead Authors	P. Hedgecock and S. Byrne (E10.02); G. Guthrie (E10.05)
Participants	A. Lowe, N. Randall, T. Mager, C. Serpan, S. Anderson, C. Whitmarsh, G. Cavanaugh, G. Martin, E. Norris, F. Stallmann, J. Perrin, and W. McElroy
Status	Being balloted at Society level, designated as E900-82. This standard is expected to be revised to provide new trend curves based on LWR power plant surveillance results; i.e., only power reactor data will be used to establish the curves that will be recommended for assessing and controlling the condition of pressure vessels for BWR and PWR nuclear power plants. Information on existing NRC-MPC-EPRI-ASTM and other metallurgical data bases is provided in References 6-9, 62, 91, and 92. Information on reevaluated exposure parameter values (flux and fluence: total, $E > 1.0$ MeV; and dpa) for 41 PWR Power plant surveillance capsules are provided in Reference 40; see Sections 2.4.1, 3.4.1, Figures 3.13, 3.14, 3.15, and Table 3.4.

E706(IIIA) Analysis of Radiometric Monitors for Reactor Vessel Surveillance

Lead Authors	L. Kellogg, F. Ruddy, W. Matsumoto, and W. Zimmer (E10.05)
Participants	G. Martin, E. Lippincott, H. Tourwé, A. Fabry, W. Schneider, A. Fudge, N. Nakazawa, Z. Sekiguchi, S. Hegedus, W. McElroy, E. McGarry, J. Grundl, W. Zijp, R. Dierckx, J. Rogers, S. Anderson, G. Cavanaugh, C. Whitmarsh, C. Cogburn, J. Williams, and F. Stallmann
Status	This standard will be balloted at E10.05 level in CY 1983. This and the E706(IIIB), E706(IIIC), and E706(IIID) method standards are being revised to include the use and reporting of equivalent fission fluxes and fluences for radiometric (RM), solid state track recorder (SSTR) helium accumulation fluence monitors (HAFM), and

damage monitors (DM). Results of the testing and verification of the procedures, data, and accuracy of RM results being obtained by service laboratories in the US and Europe are presented in Section 3.0 of this annual report, see Tables 3.6, 3.7 and 3.8.

E706(IIIB) Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Vessel Surveillance

Lead Authors	R. Gold, F. Ruddy and J. Roberts (E10.05)
Participant	L. Kellogg, E. Lippincott, W. McElroy, E. McGarry, and J. Grundl
Status	Received society approval last fall, now designated as E854-81 and appears in the 1982 Annual Book of Standards. Is currently being updated to include equivalent fission fluxes, a discussion of perturbation and position uncertainties, and the newly measured value of optical efficiency. The increased application of SSTR, RM, HAFM, and DM sensors for in- and ex-vessel physics-dosimetry surveillance programs in support of the determination of the effects of old and new fuel management schemes on the present and end-of-life condition of pressure vessels and their support structures is discussed in Sections 2.0 and 3.0.

A key development here is that the Hanford Optical Track Scanner (HOTS)* for SSTR track counting has been debugged and is undergoing calibration testing. Preliminary evaluation reveals an uncertainty range of roughly 2-3% (1σ) is being obtained. The routine application of the SSTR method for LWR in- and ex-vessel surveillance programs would not be possible without the availability of the HOTS and the more advanced Scanning Electron Microscope (SEM) automated counting systems. Development, testing, and calibration of the SEM system is not as advanced, but is still continuing. Both of these systems will be needed for the analysis of the large number of SSTR sensors irradiated or planned for irradiation in LWR-PV-SDIP neutron fields; i.e., PCA, PSF, VENUS, NESDIP, Standard NBS and CEN/SCK fields, and commercial power plants; see Table 3.9 for the commercial power plant benchmark field applications.

E706(IIIC) Application and Analysis of Helium Accumulation Fluence Monitors (HAFM) for Reactor Vessel Surveillance

Lead Authors	H. Farrar and B. Oliver (E10.05)
Participants	A. Fudge, E. Lippincott, L. Kellogg, W. McElroy, W. Hopkins, S. Anderson, H. Tourwé, A. Fabry, E. McGarry, and J. Grundl

*In Reference 52, the previous name used for this system was the Automated Optical Track Scanner (ATOS).

Status — Currently being balloted at the ASTM Society Level, and designated E910-82. As a key development here, for Point Beach 2 Surveillance Capsule R, see Table 3.9, Co/Al and Cu dosimeters were shipped to Rockwell International (RI) for He analysis. Direct comparison of the measured total He generated from the two Cu samples with the HEDL $^{63}Cu(n,\alpha)$ radiometric measurements showed excellent relative agreement with the He analysis being $\sim$8% higher. This difference is very close to that expected from the additional He production from the ^{65}Cu isotope and demonstrates that the HAFM technique can be offered as an excellent supplement, along with the SSTR and DM methods for physics-dosimetry surveillance programs. Another interesting aspect of this work, is that while some correlation work is yet to be done, the preliminary helium results indicate that the Co impurity content of the Cu dosimeters being used in Westinghouse surveillance capsules is minimal and the measurement of He may indeed be a good method for postirradiation QA confirmation of the Cu material for any surveillance capsule. As for RM and SSTR sensors, Table 3.9 provides a summary listing of the present and planned applications for HAFM sensors in commercial power plant benchmark fields.

E706(IIID) Application and Analysis of Damage Monitors (DM) for Reactor Vessel Surveillance

Lead Authors — A. Fudge, A. Fabry, and G. Guthrie (E10.05)
Participants — L. Kellogg, W. McElroy, P. Hedgecock, R. Gold, F. Ruddy, J. Roberts, M. Austin, and A. Thomas
Status — Draft outline submitted. The first draft of this standard has yet to be prepared and it is expected to concentrate on the initial use of sapphire and surveillance capsule steel correlation monitor materials. This and other candidate sensor materials for test and power reactor applications are discussed in References 18, 29, 47-50.

E706(IIIE) Application and Analysis of Temperature Monitors for Reactor Vessel Surveillance

Lead Authors — B. Seidel (E10.02) and G. Guthrie (E10.05)
Participants — A. Fabry, P. Hedgecock, A. Fudge, M. Austin, A. Thomas, G. Minsart, W. Schneider, J. Mason, A. Lowe, T. Mager, S. Anderson, G. Cavanaugh, C. Whitmarsh, W. McElroy, G. Martin, R. Gold, F. Ruddy, J. Roberts and F. Kam
Status — Draft outline to be submitted. A decision on the need and contents of this standard has yet to be made.

2.1.2 Program Documentation

The following list of planned NRC NUREG reports is provided for reference purposes. These documents are expected to be completed during the period September 1982 to September 1986 with subsequent annual updating of the loose leaf documents, as required. More detailed information on the contents of each report is provided in the minutes of the 10th LWR-PV-SDIP meeting held at NBS, October 11-15, 1982.

2.1.2.1 NUREG Report #1 (Issue Date: April 1983)
LWR-PV Surveillance Dosimetry Improvement Program:
PCA Dosimetry in Support of the PSF Physics-Dosimetry-Metallurgy Experiments
(4/12, 4/12 + SSC configurations and update of 8/7 and 12/13 configurations) W. N. McElroy, Editor

This document will provide reference physics-dosimetry information needed to support the analysis of the PSF metallurgical experiments. It will also provide updated and supplemental data in support of the previous publication: "PCA Experiments and Blind Test," NUREG/CR-1861, HEDL-TME 80-87, July 1981; see Sections 2.2 and 3.4.3.2.

2.1.2.2 NUREG Report #2 (Issue Date: March 1984)
LWR-PV Surveillance Dosimetry Improvement Program:
PSF Physics- Dosimetry-Metallurgy Experiments

Part I - PSF Physics-Dosimetry Characterization Program
W. N. McElroy and F. B. K. Kam, Editors

This document will provide reference startup physics-dosimetry information in support of the PSF metallurgical experiments; see Sections 2.3 and 3.4.3.2.

2.1.2.3 NUREG Report #3 (Issue Date: June 1984)
LWR-PV Surveillance Dosimetry Improvement Program:
PSF Physics-Dosimetry-Metallurgy Experiments

Part II - PSF Simulated Surveillance Capsule (SSC) Metallurgical Program
W. N. McElroy and F. B. K. Kam, Editors

This document will provide reference metallurgical information on measured property changes in a number of different pressure vessel and reference steels for a simulated surveillance capsule (SSC) location for two different neutron exposures of $\sim 2 \times 10^{19}$ and $\sim 4 \times 10^{19}$ n/cm^2 (E > 1.0 MeV); i.e., for tests SSC-1, and SSC-2, respectively; see Sections 2.3 and 3.4.3.2.

2.1.2.4 NUREG Report #4 (Issue Date: February 1983)
LWR-PV Surveillance Dosimetry Improvement Program:
LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium
W. N. McElroy, Editor

This loose-leaf document will provide new and/or reevaluated exposure parameter values (fluence E >1.0 MeV, dpa, etc.) for individual surveillance capsules removed from operating PWR and BWR power plants -- all in support of the development and applications of the NRC-MPC-EPRI-ASTM metallurgical data bases. The document will be revised annually as information in new and old surveillance reports is reevaluated with the FERRET-SAND and other developed methodologies; see Sections 2.4 and 3.4.1.

2.1.2.5 NUREG Report #5 (Issue Date: March 1985)
LWR-PV Surveillance Dosimetry Improvement Program:
PSF Physics- Dosimetry-Metallurgy Experiments

Part III - PSF Simulated Pressure Vessel Capsule (SPVC) and Simulated Void Box Capsule (SVBC) Physics-Dosimetery Program
W. N. McElroy and F. B. K. Kam, Editors

This document will provide reference in-situ physics-dosimetry information in support of the PSF metallurgical experiments; see Sections 2.3 and 3.4.3.2.

2.1.2.6 NUREG Report #6 (Issue Date: June 1985)
LWR-PV Surveillance Dosimetry Improvement Program:
PSF Physics- Dosimetry-Metallurgy Experiments

Part IV - PSF Simulated Pressure Vessel Capsule (SPVC) and Simulated Void Box Capsule (SVBC) Metallurgy Program
W. N. McElroy and F. B. K. Kam, Editors

This document will provide reference metallurgical information on measured property changes in a number of different pressure vessel and reference steels for simulated PV locations at the inner surface, 1/4 T and 1/2 T positions of a PWR PV wall mockup. The corresponding neutron exposures for the 2 year irradiation are $\sim 4 \times 10^{19}$, $\sim 2 \times 10^{19}$, and $\sim 1 \times 10^{19} n/cm^2$, respectively, for a $\sim$550°F irradiation temperature.

This document will also provide reference metallurgical information on measured property changes in a number of different pressure vessel support structure and reference steels for a simulated ex-vessel cavity neutron exposure of $\sim 5 \times 10^{17}$ n/cm^2 (E > 1.0 MeV), for a $\sim$95°F irradiation temperature;* see Sections 2.3 and 3.4.3.2.

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*This estimate is based on preliminary ORNL calculations, as yet unsubstantiated by measurements.

2.1.2.7 NUREG Report #7 (Issue Date: September 1984)
LWR-PV Surveillance Dosimetry Improvement Program:
PSF Surveillance Dosimetry Measurement Facility (SDMF)
W. N. McElroy, F. B. K. Kam, J. Grundl, E. D. McGarry, Editors

This will be a loose-leaf volume of results to certify the accuracy of exposure parameter and perturbation effects for surveillance capsules removed from PWR and BWR power plants. It will be updated periodically, as required; see Sections 2.3.1.2, 2.3.2.2, 3.4.2, and 3.4.3.2.

2.1.2.8 NUREG Report #8 (Issue Date: September 1985)
LWR-PV Surveillance Dosimetry Improvement Program:
LWR Test Reactor Physics-Dosimetry Data Base Compendium
W. N. McElroy and F. B. K. Kam, Editors

This will be a loose-leaf volume of results from FERRET-SAND, LSL, and other least square type code analyses of physics-dosimetry for US (BSR, PSF, SUNY-NSTF [Buffalo], Virginia, etc.), UK (DIDO, HERALD, etc.), Belgium (BR-2, etc.), France (Melusine, etc.), Germany (FRJ1, FRJ2, etc.), and other participating countries. It will provide needed and consistent exposure parameter values (fluence $E > 1.0$ MeV, dpa, etc.) and uncertainties for correlating test reactor property change data with that obtained from PWR and BWR power plant surveillance capsules. That is, with data from NUREG Report #4, these two reports will serve as a reference physics-dosimetry data base for the correlation and application of power and research reactor derived steel irradiation effects data; see Sections 2.4.2 and 3.4.3.2 and References 6-9, 62, 65, 74, 75, and 91-93.

2.1.2.9* NUREG Report #9 (Issue Date: September 1983)
LWR-PV Surveillance Dosimetry Improvement Program:
VENUS PWR Core Source and Azimuthal Lead Factor Experiments and Calculational Tests
A. Fabry, W. N. McElroy, and E. D. McGarry, Editors

This document will provide VENUS-derived reference physics-dosimetry information on active, passive, and calculational dosimetry studies involving CEN/SCK, HEDL, NBS, ORNL, and other LWR program participants; see Sections 2.4.3.3 and 3.4.3.3.

2.1.2.10* NUREG Report #10 (Issue Date: April 1984)
LWR-PV Surveillance Dosimetry Improvement Program:
NESDIP PWR Cavity and Azimuthal Lead Factor Experiments and Calculational Tests
J. Butler, M. Austin, A. Fudge, and W. N. McElroy, Editors

This document will provide NESDIP-derived reference physics-dosimetry information on active, passive, and calculational dosimetry studies involving Winfrith, CEN/SCK, HEDL, NBS, and other LWR program participants; see Sections 2.4.3.3 and 3.4.3.3.

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*To be issued by CEN/SCK and Winfrith-RR&A, with assigned NUREG Report numbers to associate them with the other 10 reports.

2.1.2.11* NUREG Report #11 (Issue Date: September 1983)
LWR-PV Surveillance Dosimetry Improvement Program:
PSF Simulated Surveillance Capsule (SSC) Results-CEN/SCK/MEA
A. Fabry and R. Hawthorne, Authors.

This dosument will provide CEN/SCK/MEA metallurgical information and results for the Mol, Belgium, PV steel irradiated in the SSC position for the ORR-PSF physics-dosimetry-metallurgy experiments; see Sections 2.3 and 3.4.3.2.

2.1.2.12 NUREG Report #12 (Issue Date: September 1986)
LWR-PV Surveillance Dosimetry Improvement Program:
LWR Test Reactor Irradiated Nuclear Pressure Vessel and Support Structure Steel Data Base Compendium
W. N. McElroy and F. B. K. Kam, Editors

This will be a loose-leaf volume of information and results for selected metallurgical experiments performed in the US (BSR, PSF, SUNY-NSTF [Buffalo], Virginia, etc.), UK (DIDO, HERALD, etc.), Belgium (BR-2, etc.), France (Melusine, etc.), Germany (FRJ1, FRJ2, etc.), and other participating countries. It will provide needed and consistent Charpy, upper shelf energy, tensile, compression, hardness, etc., property change values and uncertainties. These metallurgical data will be combined with the corresponding NUREG Report #8 physics-dosimetry data to provide 1) a more precisely defined and representative research reactor physics-dosimetry-metallurgy data base, 2) a better understanding of the mechanisms causing neutron damage, and 3) tested and verified exposure data and physical damage correlation models; all of which are needed to support the preparation and acceptance of the ASTM E706(IE) Damage Correlation and ASTM E706(IIF) ΔNDTT With Fluence Standard; see Sections 2.0, 3.0 and 5.0 and References 6-9, 62, 65, 74, 75, and 91-93.

*To be issued by CEN/SCK with an assigned NUREG Report number to associate it with the other 11 reports.

2.2 LWR PHYSICS-DOSIMETRY TESTING IN THE ORNL POOL CRITICAL ASSEMBLY PRESSURE VESSEL BENCHMARK FACILITY (ORNL-PCA)

[Variables Studied: 1) Plant dimensions - Core Edge to Surveillance to Vessel Wall to Support Structures Positions; 2) Core Power Distribution; 3) Reactor Physics Computations; 4) Selection of Neutron Exposure Units; 5) Neutron Spectral Effects; and 6) Dosimetry Measurements.]

The ORNL-PCA Pressure Vessel Benchmark Facility, Figures 2.1 and 2.2, is being used primarily in support of the development and validation of the following ASTM Standards, see Figures 3.10 and 3.11.

- Analysis and Interpretation of Nuclear Reactor Surveillance Results (IA)
- Surveillance Test Results Extrapolation (IC)
- Damage Correlation for Reactor Vessel Surveillance (IE)
- Surveillance Tests for Nuclear Reactor Vessels (IF)
- Surveillance Tests for Nuclear Reactor Support Structures (IG)
- Application of Neutron Spectrum Adjustment Methods, (IIA)
- Application of Neutron Transport Methods (IID)
- Benchmark Testing of Reactor Vessel Dosimetry (IIE)
- Correlation of ΔNDTT with Fluence (IIF)

Results of studies completed to date indicate that routine LWR power plant calculations of flux, fluence and spectrum, using current S_n transport methods can be as accurate as ±15% (1σ) for a criterion of E > 1.0 MeV if properly modeled and subjected to benchmark neutron field validation. Otherwise, errors can be a factor of two or more,[23] see Section 3.4 for additional and more updated information.

2.2.1 Experimental Program

PCA Active and Passive dosimetry measurements were made during the period from late October to early December 1981. Briefly, the emphasis of the passive dosimetry measurements dealt with 1) obtaining HEDL nuclear research emulsion measurements in the 8/7 and 12/13 configurations in the PCA pressure vessel mockup, obtaining HEDL-SSTR and -RM measurements to fill in and supplement former (1979-1981) measurements in the 8/7 and 12/13 configurations, 3) obtaining HEDL active gamma spectrometry measurements with the Janus probe in the 8/7, 12/13, and 4/12 + SSC configurations as well as measurements of the perturbation effects of the probe with a miniature HEDL ionization chamber, 4) obtaining CEN/SCK Si damage monitor measurements in the 8/7 configuration, 5) NBS confirmation of the repositioning of the PV block for the 4/12 + SSC measurements, and 6) making NBS power and run-to-run normalization monitor measurements.

2.2.1.1 PCA Passive Dosimetry Measurements

Exposures of HEDL nuclear research emulsions (NRE) in the PCA 8/7 and 12/13 configurations were successfully completed. HEDL SSTRs were exposed in a number of PCA locations of the 8/7 and 12/13 configurations. Separate SSTR runs were carried out using ^{238}U, and ^{237}Np. Axial distribution information was also obtained in the three water positions of the PCA 12/13 configuration. Good progress was made in the processing and analysis of both the NRE and SSTRs for this last set of PCA measurements and the results were used in the preparation of papers for the 4th ASTM-Euratom Symposium.[51,52]

Previous comparisons of theoretical calculations to experimental estimates lead to calculated results that are systematically lower than experiment for both ^{238}U and ^{237}Np.[23] If the present and lower SSTR experimental values being obtained for these two fission reactions are compared to the previous calculations, calculated to experimental ratios closer to unity would result.[51] These higher C/E ratios would also be more consistent with the ratios being obtained for other dosimeter reactions.[23] It is noted that in Reference 51, the present HEDL SSTR U^{238} and ^{237}Np fission rates (on the average) were observed to be about 6% lower than previously reported NBS fission chamber results. The actual difference is expected to be even higher, however, on the basis of new HEDL measurements of the optical efficiency for mica, and may reach a value near 10%. Some information on HEDL-NBS efforts to resolve these differences is provided in Section 2.4.3.

A number of HEDL radiometric (RM) sensors were also exposed in selected positions for the 12/13 configuration to complete the matrix of available RM and SSTR data for the PCA experiments.[23] Except for final counting system calibrations, the data analysis is essentially complete. The counting and analysis are a joint HEDL effort with ORNL and CEN/SCK.

Six silicon damage monitors were irradiated by HEDL in the 8/7 configuration for CEN/SCK and the sensors were shipped to Mol for analysis.[23,50]

2.2.1.2 PCA Active Dosimetry Measurements

HEDL Active Gamma Spectrometry was successfully carried out in the 8/7, 12/13, and 4/12 + SSC configurations.[25] The perturbations of the Janus probe were measured with a miniature HEDL ionization chamber.

Preliminary analysis of the Si(Li)-gamma-ray dose measurements for the 4/12 + SSC configuration indicates agreement, within experimental error, with thermoluminescent dosimeter (TLD) measurements performed by Mol in 1980.[25] The analysis of these new gamma results is continuing and will be completed in early CY 1983.

2.2.1.3 NBS Fission Chamber Measurements

Prior to the fall 1981 PCA passive and active dosimetry measurements, the 4/12 configuration of the PV mockup including the simulated surveillance capsule (SSC) was reassembled. The NBS fission chamber was again operated in the 1/4 T position at 10 kW to confirm the correct repositioning of the mockup and to compare any further PCA measurements with those made between September 1979 and November 1981.[23] Tables 2.1 and 2.2 provide a summary of results for the mentioned time period. Table 2.1 examines the reproducibility of ^{237}Np and ^{238}U fission rates in the steel block of the pressure vessel mockup at the nominally reported reactor power. Table 2.2 reexamines the same data after it was adjusted for power level differences, as given in Table 2.3. About a factor of three improvement in reproducibility is observed.

2.2.1.4 Power Level Determination and Fall 1981 Power Level Monitoring

Run-to-run power level monitoring with an NBS fission chamber[23] and an absolute redetermination of the Pool Critical Assembly (PCA) Reactor power were both accomplished by NBS at the start of FY 1982 in support of final HEDL dosimetry measurements. Table 2.3 summarizes the NBS results.

2.2.2 Calculational Program

The coupled neutron-gamma transport calculations of the PCA 12/13 configuration have been completed and documented.[76] The calculational results, Tables 2.4-2.7, confirm, in general, the revised coupled calculations of G. Minsart of CEN/SCK, Belgium. The ORNL calculations used the 47-neutron and 20-gamma group SAILWR cross-section library[77] in which the thermal neutron group cross sections were corrected for the effects of upscattering. The present calculations, which use a coarser neutron group structure than those performed two years ago,[78] reproduce:

- Very well the earlier reaction rate results ($\sim$3%), Table 2.4.
- Bare and cadmium-covered ^{235}U (n,f) fission rates at all locations where measurements were made, Table 2.5.
- Photofission enhancement effects predicted by Minsart (except for the 3/4 T location), Table 2.6.
- Gamma to neutron flux ratios of Minsart (except for the 3/4 T location), Table 2.7.

Since the present calculations of ORNL agree with the measured bare and cadmium-covered fission rates of ^{235}U (n,f) and the Minsart calculations do not, the ORNL calculations will be used until such time as the results of HEDL's final gamma-ray measurements become available for more detailed comparisons.[25]

The neutron transport calculations by Minsart for the PCA 8/7, 12/13, 4/12, and 4/12 + SSC[23,41] configurations will be made available for the NBS "Compendium of Benchmark Neutron Fields for Reactor Dosimetry," which is currently being evaluated and compiled at the National Bureau of Standards, see the E706 (IIE) Standard discussion, Section 2.1.1.

A three-dimensional Monte Carlo calculation using the MORSE code[79] was performed by ORNL to validate the flux synthesis method adopted by ORNL in their discrete ordinate analysis of measurements in the PCA. This study concludes that the flux synthesis method was sufficiently accurate that existing discrepancies between calculations and measurements lie elsewhere.[80]

To meet regulatory requirements, spectral adjustment codes must not only provide reliable estimates for spectral and exposure parameters, but must be able to determine the uncertainties associated with these parameters. A computer code, LSL,[81] which uses the least square principle was developed to determine estimates and uncertainties. The LSL program has been tested in several applications, the PCA,[23] the REAL-80,[82] and the BSR-HSST metallurgical experiments.[83,84] ORNL LSL code results have been compared with those obtained by HEDL with the FERRET-SAND code.[23] Both codes give comparable results and further comparisons of results will be obtained and reported in NUREG Report #1, see Section 2.2.3. Also planned are provisions for simultaneous adjustment of several correlated neutron spectra and for the extrapolation of spectra not accessible to dosimetry as done in Reference 58. It is noted that the simultaneous adjustment of calculated and measured spectra (and integral data) has been accomplished by HEDL with the FERRET-SAND code system for the PCA Experiments and Blind Test.[24]

2.2.3 Documentation

NUREG Report #1 of Section 2.1.2 on the "PCA Dosimetry in Support of the PSF Physics-Dosimetry-Metallurgy Experiments," which updates the information presented in Reference 23 and incorporates the data from the new PCA physics-dosimetry experiments and calculations (for the 8/7, 12/13, 4/12 and 4/12 + SCC configurations) is scheduled for completion in April 1983.

2.3 LWR STEEL PHYSICS-DOSIMETRY-METALLURGY TESTING IN THE ORR-PSF, ORR-SDMF, BSR-HSST, SUNY-NSTF FACILITIES

[Variables Studied: 1) Steel Chemical Composition and Microstructure; 2) Steel Irradiation Temperature; 3) Reactor Operating History; 4) Reactor Physics Computations; 5) Selection of Neutron Exposure Units; 6) Dosimetry Measurements; and 7) Neutron Spectral and Dose Rate Effects.]

The LWR Metallurgical Pressure Vessel Benchmark Facility (ORR-PSF) is being used primarily in support of the development and validation of the following ASTM Standards, see Figures 2.3 through 2.9.

- Analysis and Interpretation of Nuclear Reactor Surveillance Results (IA)
- Surveillance Dosimetry Extrapolation (IC)
- Displaced Atom (DPA) Exposure Unit (ID)
- Damage Correlation (IE)
- Surveillance Tests for Nuclear Reactor Vessels (IF)
- Surveillance Tests for Nuclear Reactor Support Structures (IG)
- Application of Neutron Spectrum Adjustment Methods (IIA)
- Sensor Set Design (IIC)
- Correlation of ΔNDTT with Fluence (IIF)
- Five Method Standard, IIIA, IIIB, IIIC, IIID, and IIIE

There are a number of metallurgical programs and studies that have been established to determine the fracture toughness and Charpy properties of irradiated materials as a function of chemistry, microstructure, and irradiation conditions. The ORR-PSF multilaboratory physics-dosimetry-metallurgy program is expected to provide key irradiation effects data, under well controlled conditions, to help in 1) the verification and calibration of exposure units and values and 2) the analysis and correlation of property change data obtained from this and other programs. Summary information on the status of the ORR-PSF and other program work is presented in Sections 2.3.1, 2.3.2, 2.3.3, 2.4.1, 2.4.2, 2.4.3, 2.4.4, and 2.4.5.

2.3.1 Experimental Program

2.3.1.1 ORR-PSF

The 2 year physics-dosimetry-metallurgy irradiation experiment in the ORR-PSF was completed June 22, 1982. The simulated pressure vessel capsule

(SPVC) and the simulated void box capsule (SVBC) were disassembled and the dosimetry sensors and metallurgical specimens were shipped to the appropriate participants. The final physics-dosimetry-metallurgy irradiation and temperature distribution data, and reactor power time history data for all these LWR-PV and support structure steel simulation experiments are documented in Ref. 76, see Sections 2.1.2 and 2.3.3.

FERRET-SAND physics-dosimetry results for SSC-1 have been provided to MEA and ORNL. These preliminary HEDL results have yet to be compared with those obtained by other participants (Belgium, UK, Germany, and US). Final exposure parameter values (fluence total, E > 1.0 MeV and dpa maps) for SSC-1 and SSC-2, SPVC, and SVBC must have the concurrence of all participants doing physics-dosimetry analysis.

Preliminary physics-dosimetry-metallurgy results from the simulated surveillance capsules (SSC-1 and SSC-2) have been reported by several participants in the program.[18,24,42,43,69] So far no surprises have been observed in the data. The documentation of metallurgical results for the SPVC and SVBC are scheduled for FY 1983 in MEA and FCC Reports, respectively. Planned documentation in NUREG reports is discussed in Sections 2.1.2 and 2.3.3.

2.3.1.2 ORR-SDMF

The experimental results of the Westinghouse-Combustion Engineering Surveillance Capsule Perturbation experiment have been reported in References 41 and 42; see Table 3.6 and Section 3.4.2.

The analysis of the data supplied by the four US vendors and two US service laboratories who participated in the experiment indicated biases as large as 60%. The problems were identified, and the spread in final values were greatly reduced. That is, relative agreement among the final results reported by four of the laboratories appear to be satisfactory (±5% for the non-fission dosimeters, and ±10% for the fission dosimeters.) Results from two of the laboratories appear to be consistently biased. These results demonstrate the on-going need for periodic counting laboratory intercalibrations.[42,43]

Additional information on this and other ORR-SDMF tests is provided in Sections 3.4.2, 3.4.3.1.2, 3.4.3.2 and in References 24, 31, and 41-43.

2.3.1.3 BSR-HSST

The metallurgical results of the 61W to 67W series have been reported in References 85 and 86 by ORNL. The original computer program for the statistical analysis has been modified and generalized to include nonlinear fitting. Additional information is provided in References 31, 46, 83, and 88 and in Section 3.4.3.2.

2.3.1.4 SUNY-NSTF

A new study (MEA-HEDL joint program) to evaluate additional binary and selected tertiary combinations of impurity elements has been initiated at the State University of New York (SUNY) Nuclear Science and Technology Facilities (NSTF) at Buffalo, N.Y.[69]

To date two of the four experiments have been irradiated but post-irradiation data are not yet available. MEA is responsible for the melts, experiment design, construction, irradiation, and the Charpy/tensile tests. HEDL is responsible for the small specimen compression and hardness tests, fractography, and computer analysis data/interactions. HEDL will also have responsibilties for the physics-dosimetry characterization program. Jointly, MEA and HEDL will determine the material matrices, physics-dosimetry, accomplish the data analysis, and write the final reports.

2.3.2 Calculational Program

2.3.2.1 ORR-PSF

The ORNL PSF startup experiment neutron transport calculations have been completed and the results have been compared with available results from measurements.[87] The results indicate a 10% bias relative to the PCA and the SDMF results. This comparison will be investigated further when more measurement results become available.

2.3.2.2 ORR-SDMF

Discrete ordinate transport calculations have been made by ORNL[87] and CEN-SCK[41] for the Westinghouse-Combustion Engineering perturbation experiment in the SDMF. Results indicated that the fluxes measured in a surveillance capsule differ significantly from those without the presence of the capsule. The dosimetry measurements were made by CEN/SCK and HEDL.[41,42] The comparisons indicate agreement on an absolute scale between measured and calculated reactions rates to within about 10% and agreement of the perturbation effect to within about 2%, see Section 3.4.2, Figures 3.16 and 3.17, and Table 3.6.

2.3.2.3 BSR-HSST

The neutron exposure parameters (fluence >1 MeV, fluence >0.1 MeV, and dpa) for the Heavy Section Steel Technology (HSST) experiments can be determined to $\pm 10\%$ (1σ) variance.[84,88] The exposure parameters were obtained by combining the measured reaction rates, the calculated neutron transport fluxes, and the cross section values in the logarithmic least squares adjustment code LSL.[81]

2.3.2.4 SUNY-NSTF

HEDL will have the lead responsibility for modeling, completing, and documenting the results for the transport calculations for the SUNY-NSTF

(Buffalo, NY) MEA-HEDL chemistry-metallurgical tests. ORNL will provide technical assistance in the use of the DOT transport code and offer suggestions as to the modeling of the core and experiment. MEA will provide detailed information on the Buffalo irradiation rigs and their operation; i.e., materials, geometries, dimensions, with tolerances, water and air gaps changes resulting from temperature control, thermocouple lead gaps, etc. The calculations are scheduled to be completed in FY 1983. HEDL will use the FERRET-SAND code to obtain dosimetry adjusted neutron exposure parameters for this important series of metallurgical irradiations. Both HEDL and MEA dosimetry measurement results will be available for input to the FERRET-SAND adjustment code.

2.3.3 Documentation

Information on the use and results of the physics-dosimetry-metallurgy calculations for the ORR-PSF measurements are scheduled for publication in five NUREG Reports (Section 2.1.2) beginning in 1983. The purpose here is to compile in one set of reference reports the most significant results of the ORR-PSF experiments and their impact on the assessment and control of the present and EOL condition of LWR pressure vessel and support structure steels.

2.4 ANALYSIS AND INTERPRETATION OF POWER REACTION SURVEILLANCE AND RESEARCH REACTOR TEST RESULTS

[Variables Studied: All those listed in Section 2.0]

A primary objective of this multilaboratory program is to help in the development of statistically valid neutron radiation embrittlement data bases (NRC-MPC-EPRI-ASTM and others)[6-9,62,65,74,75,91-93] for use in the critical evaluation of the procedures and data used for predicting the fracture toughness and embrittlement of irradiated reactor pressure vessel and support structure steels.

Analysis of the existing and new additions to these data bases (from test and power reactors) has revealed that the variance of test data does not arise entirely from material variability. A substantial portion stems from lack of consistency in the application and/or shortcomings in test methods and control of important variables associated with the "reactor systems analysis," "physics-dosimetry," "metallurgy," and "fracture mechanics" disciplines.[2,3,6-9,18,62,64,65,69,74,75,84-86,88,91-93]

Analyses of PWR surveillance capsule and research reactor data indicate that long-term LWR power plant surveillance capsule and short-term research reactor (∿288°C irradiation temperature) neutron-induced property change data for steel (base metal, heat-affected zone and weld metal) can show significantly different neutron exposure dependencies.[2,3,6-9,24,40,55,62,64,65,69-71,74,75,91-93] For instance, for low-flux surveillance capsule irradiated materials, the neutron-induced damage may increase at a rate per unit fluence similar to that of high flux test reactor irradiated materials, up to some level of exposure which appears to be a function of chemistry and microstructure. At exposures above this level, the rate of embrittlement is much reduced and it appears that the embrittlement saturates.[2,3]

The functional forms of the chemistry term A and the slope N, of the equation $\Delta NDTT = A(\phi t)^N$, are as yet not well defined, but recent studies suggest that these forms should show a Cu and Ni effect for the "A" term, with the exposure exponent "N" assumed to be either an adjustable constant, or possibly a linear function of the $\log_e(\phi t)$.[74,75] It is further concluded, at least for the present, that research reactor and surveillance capsule irradiation effects data should not be combined to predict PV steel fracture toughness and embrittlement as a function of neutron exposure without having: 1) More precisely defined and representative physics-dosimetry-metallurgy data bases, 2) a better understanding of the mechanisms causing neutron damage, and 3) tested and verified exposure data and physical damage correlation models; all of which are needed for the preparation and acceptance of the ASTM E706(IE) Damage Correlation, ASTM E706(IIF) ΔNDTT With Fluence, and other E706 standards, see Section 2.1.1.

Summary information is presented in Sections 2.4.1 through 2.4.5 on the results of recent LWR-PV-SDIP studies associated with physics-dosimetry-metallurgy data development and testing for power reactor surveillance and research reactor irradiation effects programs.

2.4.1 Surveillance Capsule Data Development and Testing

As part of the LWR Program, statistically based data correlation studies have been made by HEDL and other program participants using existing metallurgical data banks in anticipation of the analysis of new fracture toughness and embrittlement data from the BSR-HSST, SUNY-NSTF, ORR-PSF and other experiments.[3,6,7,9,18,31,40,46,62,64,65,74,75,83-85,88] Summary information and results related to these studies follow.

In References 74 and 75, information is presented on recent results in trend curve studies. There is some mention of recent results and suggestions which have originated with G. Odette, J. Varsik, G. Guthrie, and P. Randall. Varsik and Odette have done separate analyses for subsets of the existng data, separating weld data from plate data. Odette has recently suggested that the plot of log (ΔNDTT) vs log (ϕt) is not quite linear, as it would be if

$$\Delta NDTT = A\ (\phi t)^{N} \tag{1}$$

gave a perfect fit for a fixed value of N. Therefore, it appears that an improved fit could be achieved by allowing N to be a slowly varying function of ϕt; e.g.,

$$N = B + C \cdot \log_e\ (\phi t). \tag{2}$$

Varsik, Odette, and others have found that different fixed values of N are applicable for the separate data populations, with the weld data having the lowest values of the exponent. Varsik has also found that the chemical premultiplier function "A" differs for the separate populations. The importance of particular elements depends on the product type. Odette has suggested that at high ratios of Ni/Cu, the differential effect of additional nickel is not as important as it would be for low values of Ni/Cu.

It was mentioned that macroproperties of Cu-Ni alloys give a justification for this effect for the Cu-Ni clusters, which are believed to pin the dislocations in plastic deformation. The Cu-Ni systems show Ni-like properties at high Ni concentrations, but show Cu-like properties for alloys having relative Cu concentraions >65%. Therefore, at low ratios of Ni/Cu, the Ni may join the copper clusters and act as a copper extender. At higher levels of Ni/Cu, additional Ni atoms may not be able to extend the number or volume of the Cu clusters without changing their character; see also Reference 64.

Further, there has been some discussion of the effect of considering the fluence errors. When the fluence errors are included in the least squares program, the result is an increase in the calculated value of the fluence exponent N. It was pointed out that least squares fits that do not consider the fluence errors produce a value of N which is slightly in error on the low side, giving a slight over estimate of the saturation effect.

Information is also given in References 74 and 75 on recent cooperative work between Guthrie and Randall. A study of surveillance capsule data provided by NRC was used to produce a trend curve formula of the type of Eq. (1). The chemistry-dependent premuliplier "A" was a function of the Cu concentration and the product of the Cu and Ni concentrations, Cu•Ni, with a fixed exponent N for the fluence. The computer code used in the project included the fluence error in the sum of squares of residuals, and produced an exponent value of $\sim$0.27. The standard deviation was 24°F. More recent work has used a fluence exponent of the form of Eq. (2) while the chemistry-dependent premultiplier contained provision for the Ni saturation effect at high ratios of Ni/Cu. The form of the chemistry-dependent premultiplier was

$$A = \chi(1) + \chi(2) \cdot Cu + \chi(3) \cdot Cu \cdot \tanh\left[\chi(4) \cdot \frac{Ni}{Cu}\right], \tag{3}$$

where the $\chi(1) \cdots \chi(4)$ are adjustable parameters. The term involving $\chi(3)$ expands to $\chi(3) \cdot Cu \cdot \chi(4) \cdot (Ni/Cu)$ for low values of $\chi(4) \cdot (Ni/Cu)$. This then reduces to $\chi(3) \cdot \chi(4) \cdot Ni$. For high values of $\chi(4) \cdot (Ni/Cu)$, the tanh function saturates and additional Ni does not affect the calculated Charpy shift. The use of this formula, Equation (3), has resulted in a standard deviation of $\sim$20.4°F for the NRC-selected 138 surveillance capsule data points.

For the above HEDL-NRC work, the effect of using dpa as an exposure parameter as well as fluence (E > 1.0 MeV) was investigated using new HEDL reevaluated exposure values developed by Simons, see Table 3.4 and Reference 40. As expected, the use of dpa in place of fluence made essentially no difference since all of the 138 data points came from surveillance capsules where the neutron spectral shapes were quite similar. An important aspect of this work, nevertheless, was the "first" development of trend curves based on dpa for the NRC-selected data base. These curves are needed and can be used to account for the effects of the large neutron spectral shifts between surveillance capsule and in-vessel wall locations, such as at the 1/2 T and 3/4 T, positions, see Sections 3.3 and 5.0.

It should be noted that the NRC-supplied 138 point Charpy metallurgical data set had an NRC-selected set of fluence (E > 1.0 MeV) values provided by Randall. These NRC (Randall) values are compared with the new HEDL (Simons) values in Reference 75 and some significant differences are noted, but in general the differences are not too large. The NRC-selected fluence values had made use of some of the earlier FERRET-SAND code-derived exposure parameter values of Simons. The NUREG Report #4, Section 2.1.2.4, which is to be issued at the end of February 1983, will provide the most recent Simons' reevaluated results for flux (E > 0.0 MeV, Thermal, E > 0.1 MeV, E > 1.0 MeV), 53 Group a priori and adjusted fluxes, dpa/second, dpa, fluence (E > 1.0 MeV), effective full power operation time in seconds for a specified reactor power and the associated uncertainties for approximately 41 surveillance capsules.

In Equations (1) to (3), no provision has been made for neutron damage caused by low energy thermal and epithermal neutrons. For surveillance capsule PV steels irradiated at $\sim$288°C (550°F), Varsik has reported on the observation of a possibly significant low energy neutron effect. In Ref. 65, he states: "The examination of activation fluence as a neutron exposure parameter was also hindered by a low quantity of data. Results did indicate that when the influence of thermal neutrons was considered, both irradiation time and thermal neutron irradiation became highly significant in correlating transition temperature shift. These results are similar to those noted in a previous investigation and suggest the presence of a transition shift "saturation" effect associated with thermal neutrons and irradiation time."

With a low energy (thermal) to fast (E > 1.0 MeV) fluence ratio well below 10 and often near unity for PWR surveillance capsules, the relative contribution of thermal neutrons to the calculated dpa and measured Charpy Shift would appear to be small and less than a few percent, see Figure 3.25, Section 5.0 and Reference 56.

The earlier A302B steel results of Serpan et al., for irradiation temperatures less than 116°C (<240°F), supported a thermal neutron contribution to damage for research reactor test locations with high thermal to fast (E > 1.0 MeV) neutron ratios (> about 10); but the non-boron containing A533B steel results of Alberman et al., did not for an irradiation temperature of $\sim$100°C ($\sim$212°F); see References 6 and 10 of Section 5.0. Alberman, et al. did observe, however, a substantial thermal neutron effect at $\sim$100°C for iron specimens with boron concentrations up to 5 ppm, irradiated in high thermal to fast neutron flux ratios. Above the 5 ppm level, increased boron content appeared to have little influence on any increases in measured mechanical property. The boron content of the A302B steel used by Serpan et al. was estimated to be in the range of 1-6 ppm. Consequently, and depending on the boron content of the steel, the irradiation temperature and time at temperature, and the thermal flux levels encountered for individual surveillance capsules, there could be a small contribution from thermal neutrons to the observed damage in PV steels. This might also apply for some support structure steels.

If the thermal neutron effect suggested by the data used by Varsik is found to be real, a mechanism other than just displaced atoms of iron must be contributing to the damage. For instance, such a mechanism might be associated with the interaction of thermal and epithermal neutrons with the boron in PV steels at the elevated temperature of $\sim$288°C ($\sim$550°F) encountered in most operating PWR surveillance capsules. That is, while Alberman et al find less than a 1% contribution from thermal neutrons to the damage in A537 steel irradiated at 60°C (with approximately equal thermal and fast fluxes), a greater relative amount of residual neutron damage might remain (after in-situ at temperature annealing) from boron (n,α) recoils and helium production than from dpa at 288°C.

The planned HEDL-RI use of the HAFM method to determine the helium content of selected irradiated Charpy specimens from a number of PWR surveillance capsules is expected to shed some additional light on this matter; i.e., provide an estimate of the effective boron content by measurement of the helium content. The measurement of the helium produced in irradiated PV steel Charpy specimens is being accomplished to determine if scrappings from PWR pressure vessels might be used as HAFM dosimetry sensors.

The clarification of the above or other thermal neutron effects could be very important for establishing the present and EOL condition of PV steels because 1) a reduction of scatter in the existing and future surveillance capsule physics-dosimetry-metallurgy data could result and 2) any thermal neutron contribution to damage observed at an accelerated surveillance capsule location could result in either an over- or under-prediction of damage when applied to the surface, 1/4 T, 1/2 T, etc. locations within the vessel wall; i.e., since the intensities of the thermal and fast neutron components that reach and penetrate the vessel wall will be considerably lower than that at the surveillance position and their relative ratio (thermal/fast) will be changed by absorption and scattering, first in water and then in iron within the vessel wall itself.

The results of the above work are being used in studies associated with the verification of the procedures and data being recommended in the new ASTM E706(IE) Damage Correlation, ASTM E706(IIF) ΔNDTT With Fluence, and other E706 standards.

2.4.2 Research Reactor Data Development and Testing

As part of the LWR Program, statistically based (as well as other) physics-dosimetry-metallurgy data analysis and correlation studies using research reactor data are being made by ORNL, MEA, HEDL, and other program participants,[3,9,18,29,31,46,49,56,57,64,67-71,76,81-88] The reader is referred to Sections 2.3.1 and 2.3.2 and the appropriate references given above for more information on the ORNL, MEA, and HEDL studies. Here, the discussion will be limited to the consideration of recent HEDL work directed towards the reevaluation of exposure parameter values (fluence E > 1.0 MeV and dpa) for selected sets of metallurgical data obtained from research reactor tests.

As mentioned in Section 5.0, under the Section on "Supporting Evidence," Simons has reevaluated the exposure data developed by Serpan and McElroy on the shift in ductile-brittle transition temperature for five data sets on three steels using a simple correlation model suggested by Odette. Statistical analysis of fits to the data using dpa, fast fluence (E > 1.0 MeV), and three other exposure indices were performed. The relative variances for dpa and fast fluence are given in Table 2 of Section 5.0. Clearly, dpa gives a better overall correlation of the data than does fast fluence. Information on the results of other studies by Odette, Mas et al., and Alberman et al., are also given in Section 5.0 with similar conclusions.

Physics-dosimetry data from a previous NRL metallurgical irradiation conducted by Serpan in a thermal shield-PV mockup in the Industrial Research

Laboratory (IRL) pool reactor were reanalyzed with the FERRET-SAND computer code by Simons in FY 1982. The uncertainty in the exposure rates through the mockup due to uncertainties in the neutron spectrum adjustment were 10-11% for flux (E > 1 MeV) and 9-13% for displacements per atom per second (dpa/s). A comparison of the original fast flux determined by Serpan and McElroy with the new FERRET-SAND values, using the same reaction rate data, generally shows good agreement. The maximum deviation of 25% occurs at the location in front of the thermal shield. The flux varied by a factor of 44 through the thermal shield-PV mockup and the dpa/s varied by a factor of 24. The ratio of dpa/s to flux (E > 1 MeV) varied by 40% through the mockup for positions where measurable ΔNDTT were reported by Serpan. Consequently, with only a 40% change, it will be difficult to accurately assess the neutron spectrum effect on the irradiation induced metallurgical property changes measured by NRL for this IRL experiment; i.e., show with high confidence that the measured Charpy shifts and calculated dpa values have the same gradient through the simulated PV wall. It is noted that the metallurgical specimens were irradiated at the reactor pool water ambient temperature and not at $\sim$550°F. These updated IRL research reactor metallurgical results have yet to be compared with ORNL, MEA, HEDL, or other research reactor data. Together with these other data, they will be used in studies associated with the verification of the procedures and data being recommended in the new ASTM E706(IE) Damage Correlation, ASTM E706(IIF) ΔNDTT With Fluence, and other E706 standards.

2.4.3 Benchmark Referencing Program

Benchmark referencing studies on both the experimental and calculational aspects of the LWR-PV-SDIP are important program elements. The results of such studies are discussed and referenced throughout Sections 1.0, 2.0, and 3.0. In subsequent subsections, the discussions will center on current or planned benchmark referencing studies involving NBS and other program participants.

2.4.3.1 Benchmark Referencing of Solid State Track Recorders (SSTR)

In the first NUREG Report covering the 1978-1980 PCA measurements and blind test of transport calculations, discrepancies were noted between fission chamber and SSTR results.[23] Resolution of the problem was hampered by the fact that SSTR methodology had not yet been benchmark referenced. Considerable effort has now been expended to reconcile this issue, as follows:

a) HEDL-SSTR Response Intercompared to NBS-Fission Chamber Response

Prior to exposing any solid state track recorders (SSTRs), the relative fission rates of various fissionable deposits were obtained with the NBS double fission chamber. In practice, two fission chambers were used simultaneously on either side of an NBS ^{252}Cf source. This configuration reduces uncertainties in source-to-deposit distances and permits exposure of four deposits, each pair of which is in a back-to-back orientation separated only by 10 mil of stainless steel. Relative fission rates may be obtained to accuracies of several tenths of a percent in this manner.

Subsequently, SSTRs were placed against one of the deposits in each chamber and were exposed in the NBS ^{252}Cf standard neutron field. In this manner, the number of fission events which the SSTRs saw was directly monitored by an active fission chamber. A series of SSTR irradiations produced 12 track recorders for direct intercomparison of fissions as detected by SSTRs and fissions detected by the NBS fission chambers.

Finally as a quality control measure, the relative counting rates of the fissionable deposits were again compared to assure that there was no loss of deposit material because of an SSTR being in contact with a deposit.

This intercomparison was accomplished as a blind test, and results are being sent to R. Armani of Argonne National Laboratory who will serve as referee. Note that although this intercomparison is a vital step in benchmark referencing of SSTR methodology, the other important issue of mass assay accuracy was not addressed in the present comparisons.

b) Benchmark Referencing of SSTR Deposits Used for PCA Measurements

Essentially all of the HEDL ^{238}U and ^{237}Np fissionable SSTR deposits used for measurements in the pressure vessel mockup were sent to NBS and were subjected to certified fluence irradiations in the NBS ^{252}Cf standard neutron field. These calibrations required in excess of 200 hours of irradiation time to accomplish approximately 23 experiments. These exposures were completed in late FY 1982 - early FY 1983 so that the deposits could be shipped to Belgium for the scheduled VENUS measurements in January and February 1983. These NBS-HEDL irradiation experiments will 1) benchmark reference the PCA-SSTR measurements to the NBS ^{252}Cf neutron field, 2) will provide calibration factors for deriving fission equivalent fluxes at VENUS and NESDIP, and 3) will provide a certain amount of fission deposit mass assay intercomparisons.

2.4.3.2 Benchmark Referencing of Helium Accumulation Fluence Monitors (HAFM)

In cooperation with Rockwell International (RI), certified fluence irradiations are underway for a variety of Helium Accumulation Fluence Monitor (HAFM) neutron sensors. These calibrations are being carried out in the NBS Intermediate-Energy Standard Neutron Field (ISNF). Fluence monitoring is being accomplished with the $^{198}Au(n,\gamma)^{199}Au$ epithermal neutron and the ^{115m}In fast neutron reactions. The flux calibration was transferred from the NBS ^{252}Cf field by calibrating the response of a ^{237}Np fission chamber in the ^{252}Cf field and using known spectrum-averaged ^{237}Np cross sections for the ^{252}Cf and ISNF fields, respectively. RI will analyze the irradiated HAFM sensors for helium content and the results will be used to provide verification and calibration checks for the individual sensors; all in support of the preparation, acceptance, and application of the ASTM E706(IIIC) HAFM Standard.

2.4.3.3 Support of Other International Benchmark Experiments

a) Mol, Belgium VENUS Zero-Power Mockup of a PWR Core-Baffel-Barrel-Thermal Shield Configuration

HEDL and NBS will actively participate in the VENUS measurement program in January-March 1983. Plans have been made to send appropriate HEDL and NBS staff members 1) to accomplish and assist with the VENUS experimentation, 2) to make standard neutron field calibration measurements, and 3) to prepare for measurements to be made at NESDIP. In addition to the HEDL active and passive dosimetry, a series of NBS fission chambers will be involved in the various measurement sequences and a redetermination of the intercalibrations existing among the various NBS and CEN/SCK standard fission neutron fields will be accomplished.

b) United Kingdom, NESDIP Power-Reactor Ex-Vessel Cavity Benchmark Experiment

HEDL participation will involve both active and passive dosimetry measurements. NBS participation in the NESDIP experiment, as seen at the moment, involves relating the measurements in NESDIP and the flux at the center of the high-flux benchmark field, called NESSUS, in the NESTOR reactor to the intercalibration results of the NBS and CEN/SCK ^{235}U fission neutron fields and to the NBS ^{252}Cf fission neutron field.

c) ANO-2 Benchmark Experiments

Separate NBS benchmarking is in progress for radiometric foils and analyses by Arkansas University, for the ANO-2 cavity dosimetry, and Battelle/Columbus, for the ANO-2 first Surveillance capsule dosimetry, see Table 3.9. To date, certified ^{235}U fission fluence irradiations have been accomplished for the $^{58}Ni(n,p)^{58}Co$, $^{54}Fe(n,p)^{54}Mn$ and the $^{238}U(n,f)$Ba-La reactions. Both service laboratories will count these certified fluence standards.

d) PWR Surveillance Capsule Co-Al Alloy Benchmark Experiment

NBS now has Co-Al alloy specimens from HEDL for quality assurance checks of the cobalt content. The specimens are from the Point Beach 2 Surveillance Capsule R and are rather radioactive with ^{60}Co; therefore, the exact method of quality assurance is still under consideration.

2.4.4 Fourth ASTM-EURATOM International Symposium on Reactor Dosimetry

The subject symposium was held at the National Bureau of Standards, Gaithersburg, Maryland, March 22-26, 1982. This series of biennial international symposia brings together specialists from many countries to provide a forum for the exchange of new and critical information concerning the techniques and applications of neutron and gamma dosimetry in materials irradiation studies.

The theme of the symposium was radiation metrology techniques, data bases, and standardization. Application and requirements for radiation metrology of irradiated fuels and materials in fission and fusion technology were emphasized. The proceedings were compiled and edited by F. Kam of ORNL, Program Committee Chairman, and published in July 1982.[31] Publication of the Proceedings and translations during the conference were supported by the US Nuclear Regulatory Commission.

The symposium was attended by 119 participants from 15 countries. The Proceedings contain the full text of approximately 100 papers plus highlights by the chairmen of the 11 sessions and 4 workshops.

An ASTM Certificate of Recognition was presented to C. Serpan (former chairman of ASTM Subcommittee E10.05 on Dosimetry) and Ugo Farinelli (former chairman of the EURATOM Working Group on Reactor Dosimetry) for their foresight and joint efforts in establishing and strongly supporting this highly successful series of ASTM-EURATOM International Symposia on Reactor Dosimetry. In addition to ASTM and EURATOM, DOE, NRC, NBS, and EPRI were cosponsors of the Symposium with assistance from IAEA.

2.4.5 Documentation

The documentation of the results of studies related to the analysis and interpretation of power reactor surveillance and research reactor test data is expected to be accomplished in 1) individual participating laboratory technical reports, 2) LWR-PV-SDIP Quarterly and Annual Progress Reports, and 3) NUREG Reports, #4, 8, and 12; see Section 2.1.2.

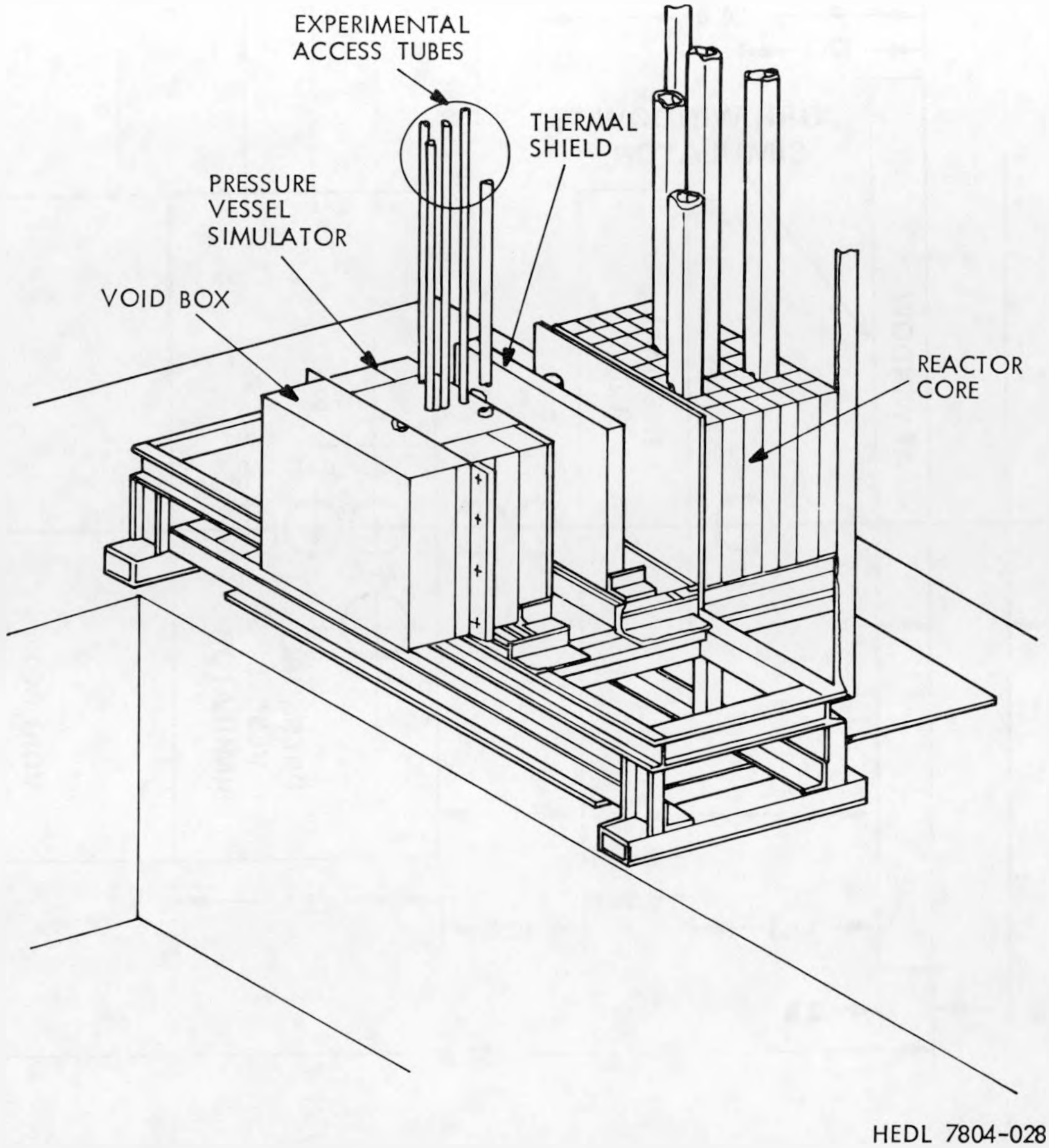
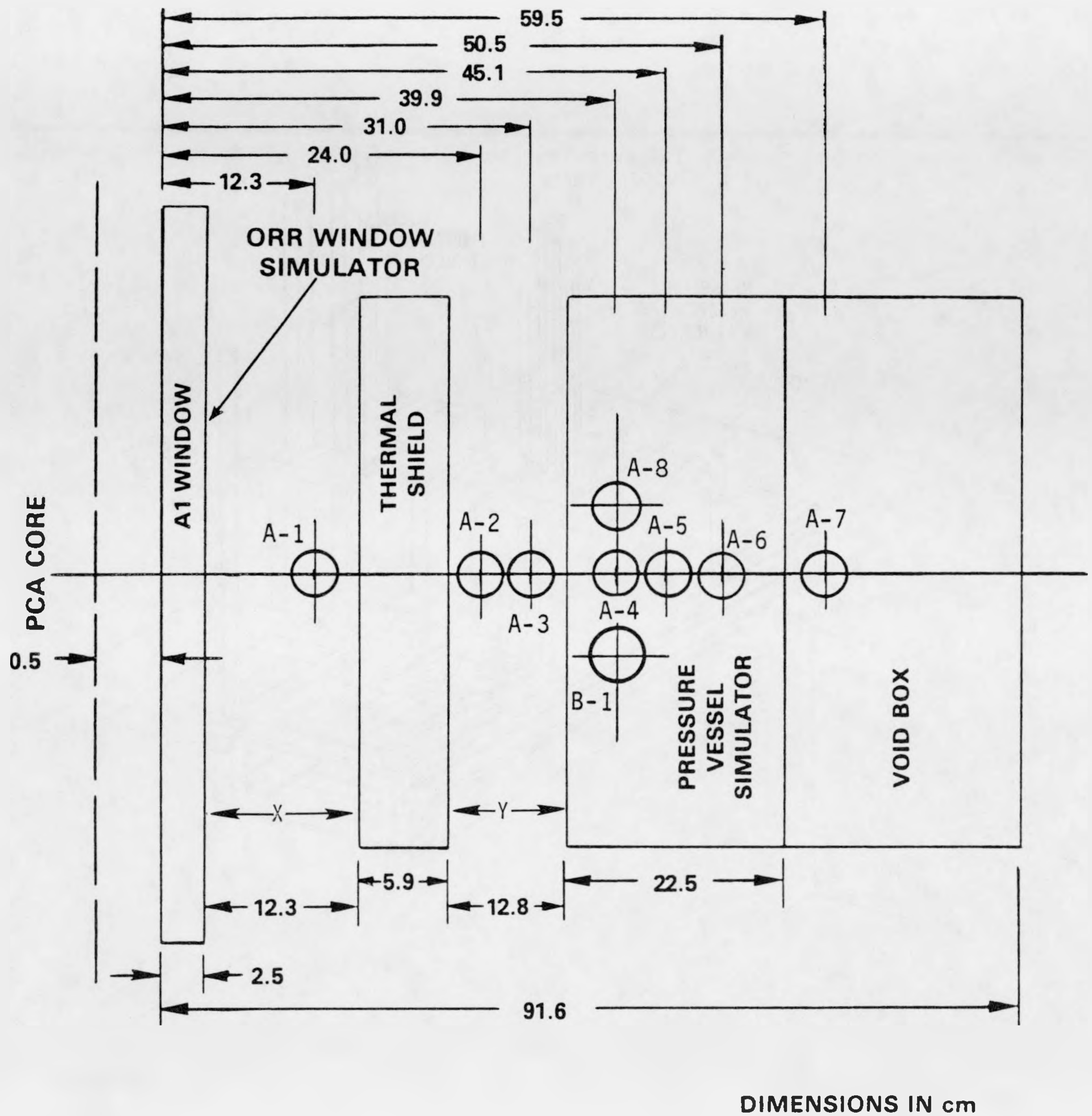


FIGURE 2.1 Pressure Vessel Wall Mockup Schematic of Two Equivalent Facilities Constructed at ORNL. The high-flux version at ORR (PSF) includes damage exposure of metallurgical test specimens; the low-flux version near a low-power critical assembly (PCA) focuses on active and passive physics-dosimetry measurements. P05788-2



– SCHEMATIC, NOT TO SCALE –

FIGURE 2.2 PCA Experimental Configuration: Locations A1, A2, A3, A4, A5, A6, A7, A8 and B1 in the PCA 12/13 Configuration. Representative for other configurations except for different X and Y dimensions.

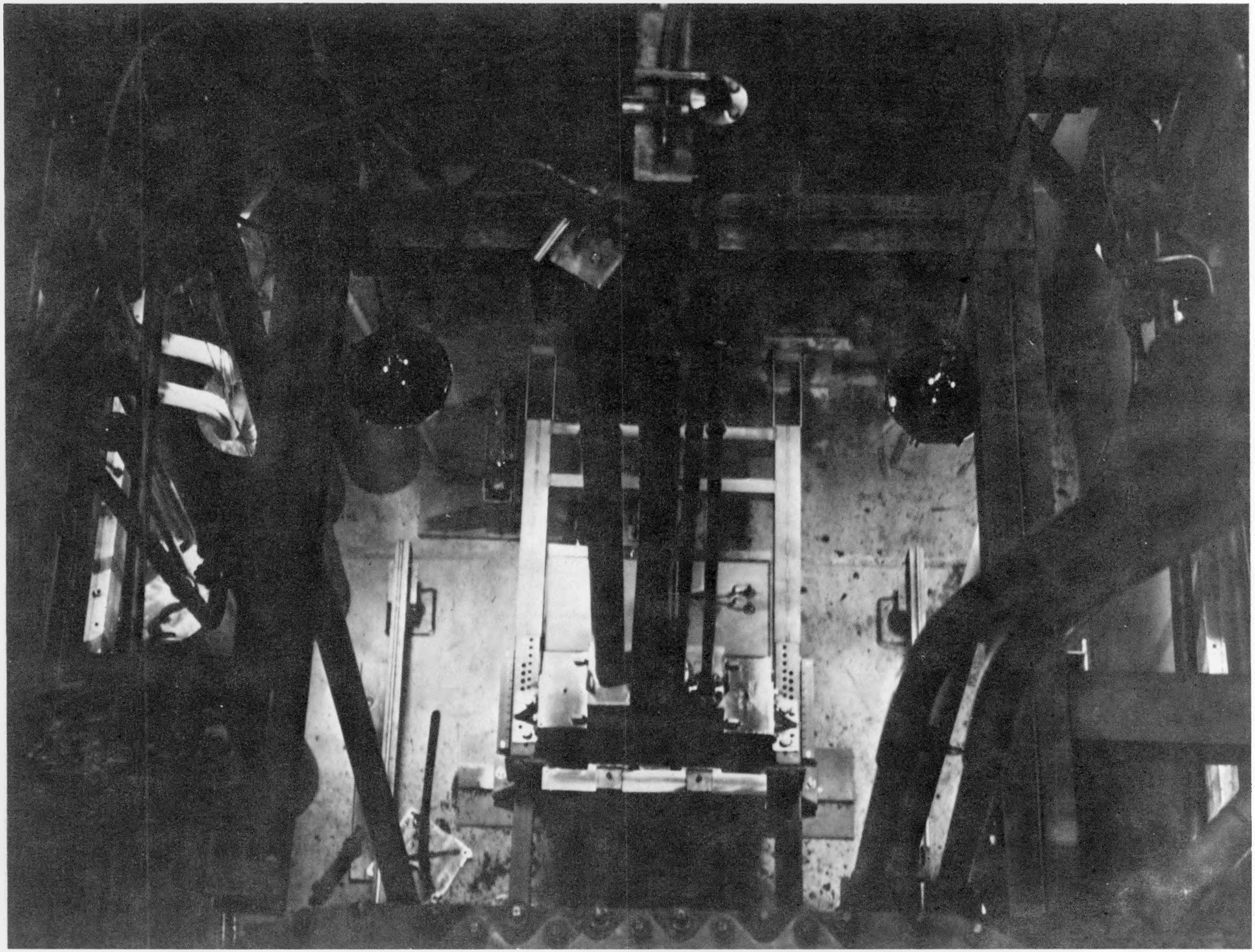


FIGURE 2.3 LWR Metallurgical Pressure Vessel Benchmark Facility. Neg 8010270-19

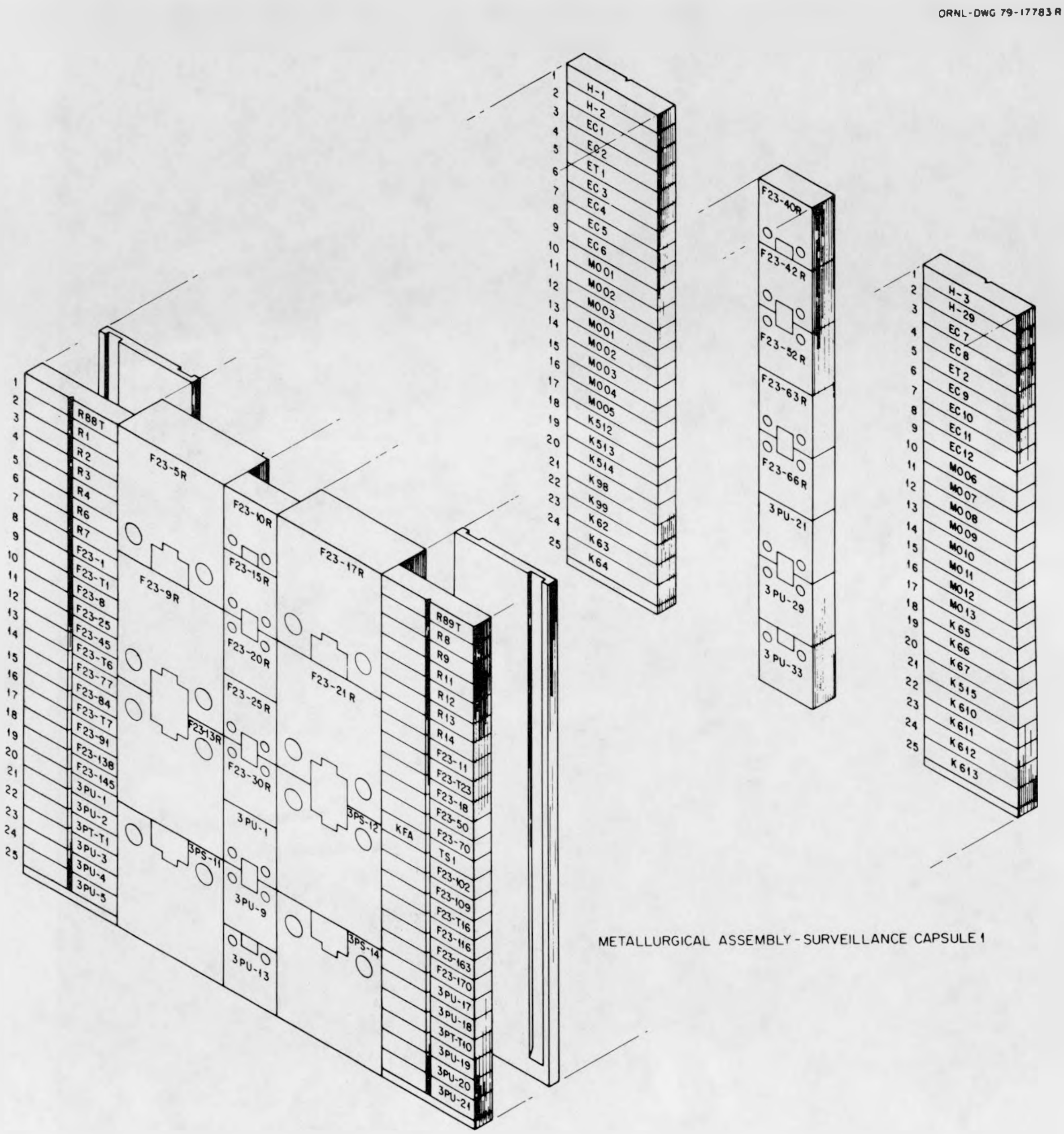


FIGURE 2.4 Metallurgical Assembly Located in the Surveillance Capsule. Neg 810270-10

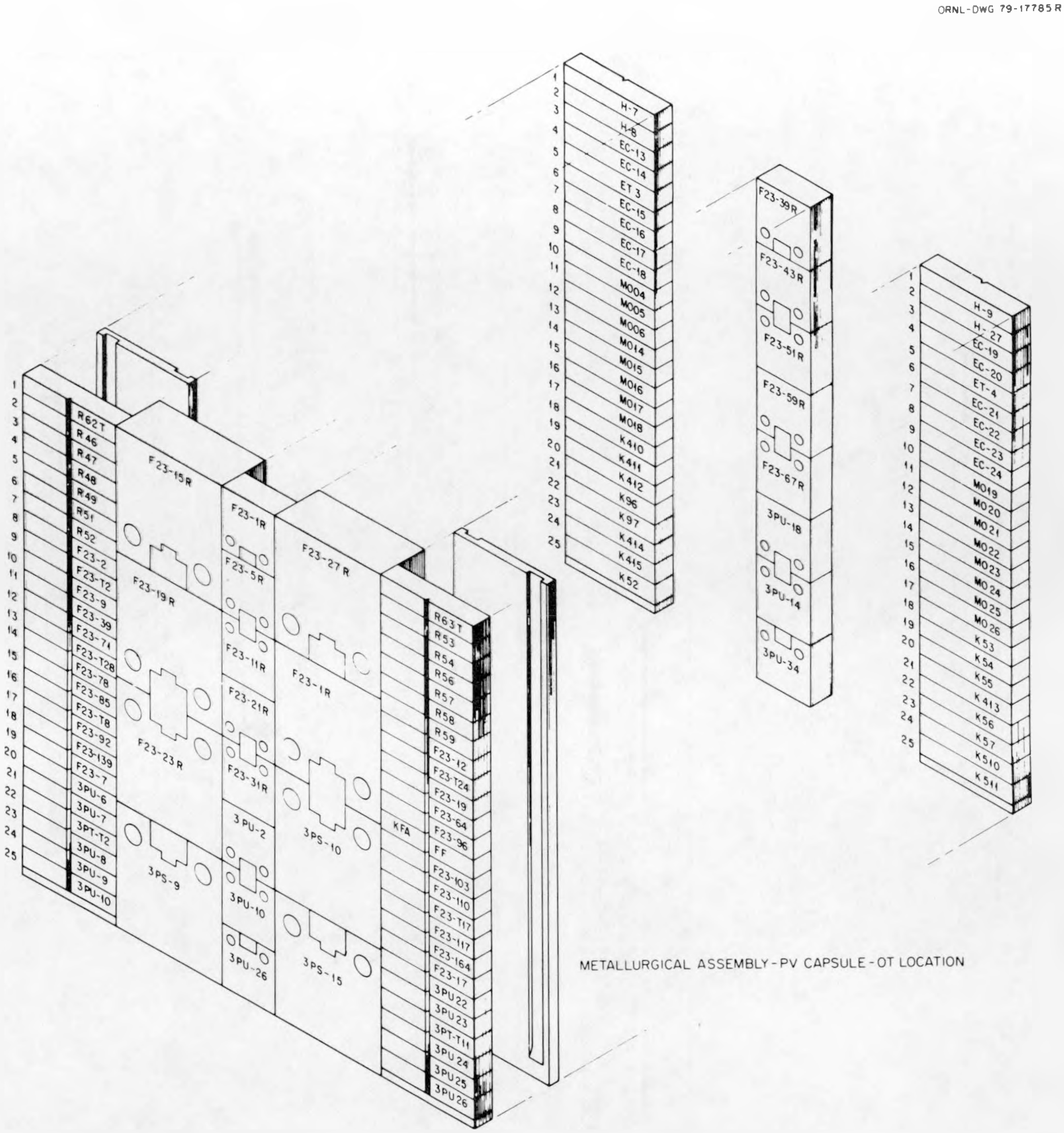


FIGURE 2.5 Metallurgical Assembly at the Inside Surface (Zero-Thickness Position, OT) of the Simulated Pressure Vessel Capsule. Neg 8010270-9

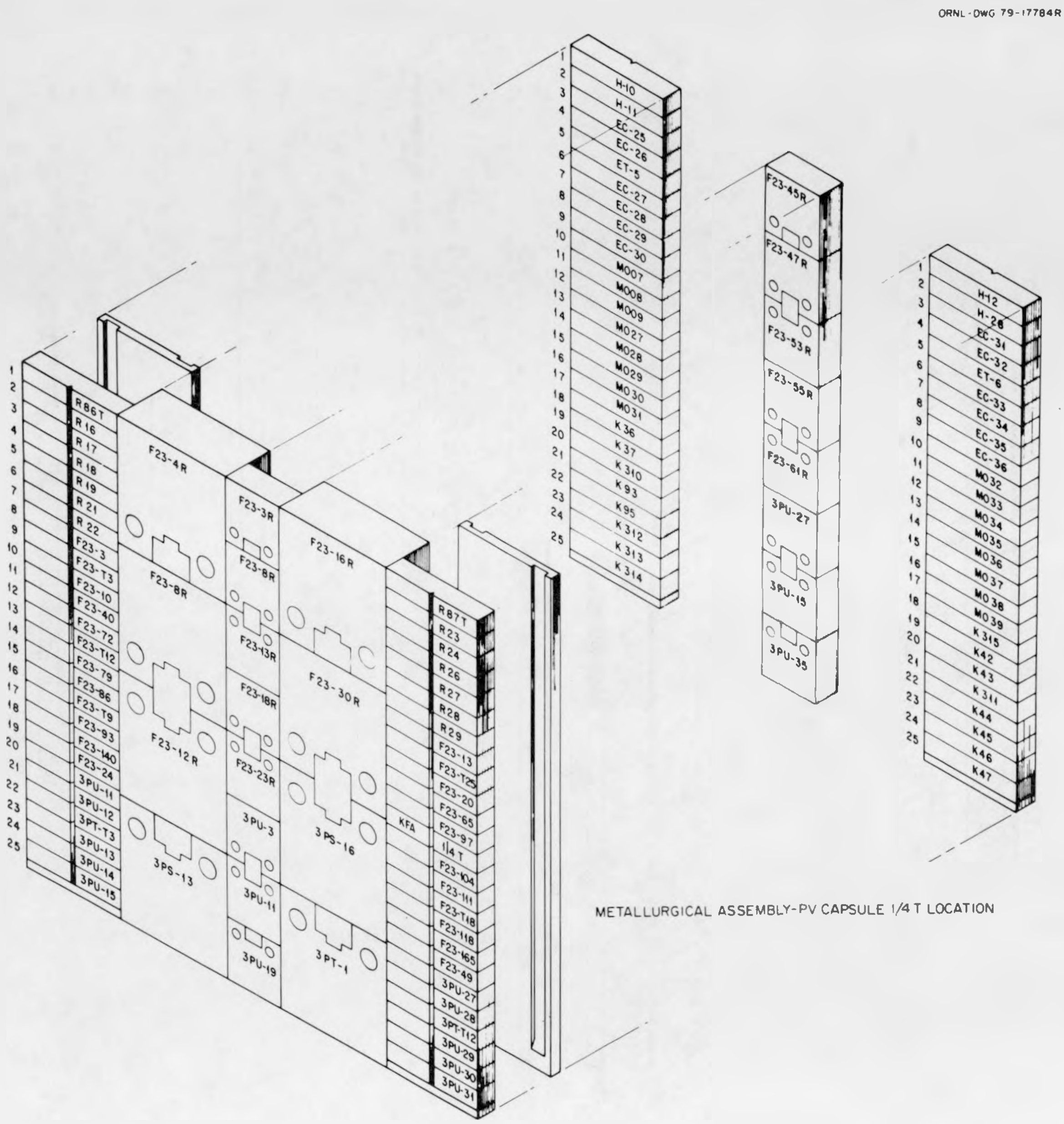


FIGURE 2.6 Metallurgical Assembly Located at the Quarter-Thickness (1/4 T) Position in the Simulated Pressure Vessel Capsule. Neg 8010270-8

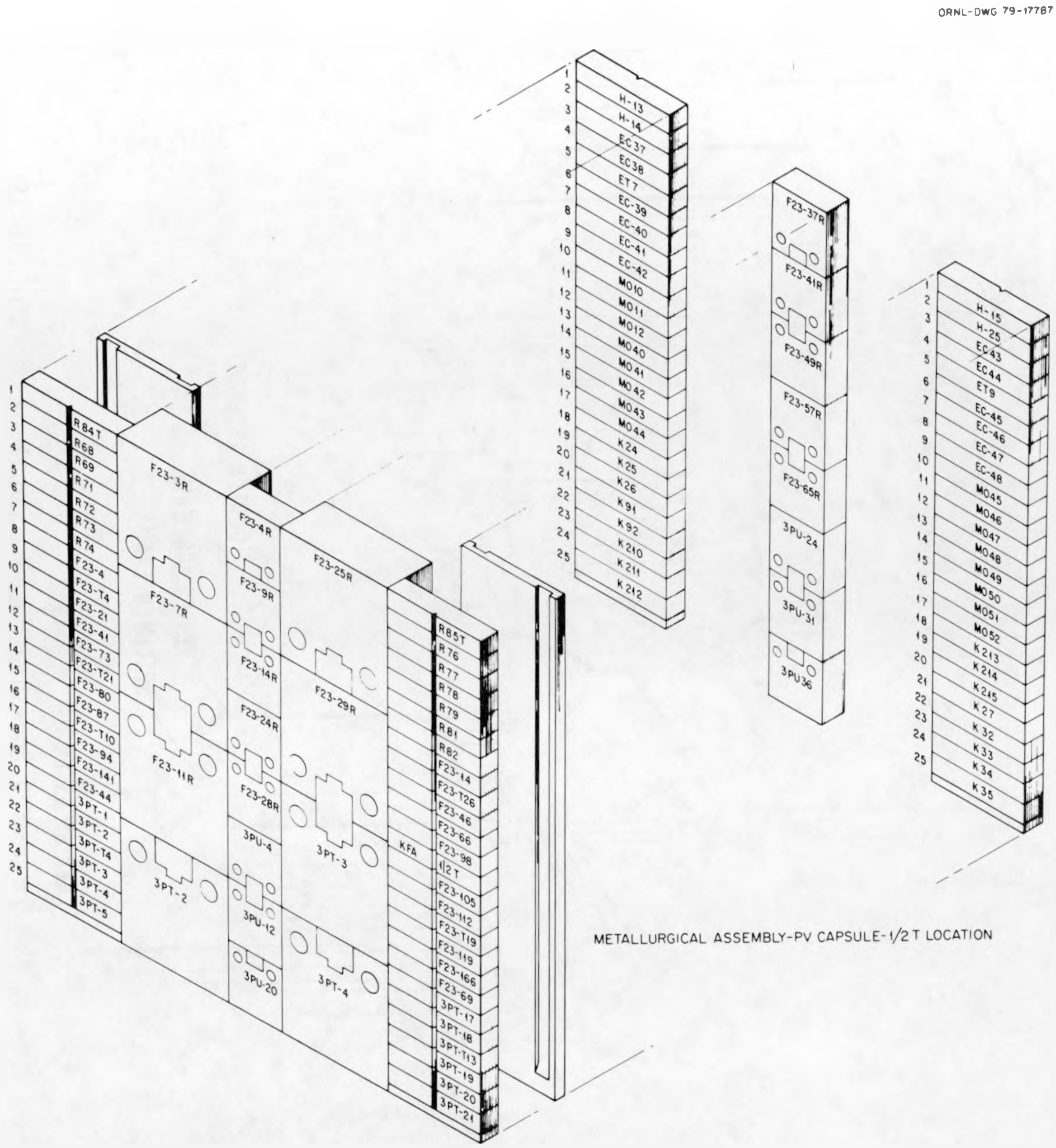


FIGURE 2.7 Metallurgical Assembly Located at the Half-Thickness (1/2 T) Position in the Simulated Pressure Vessel Capsule. Neg 8010270-7

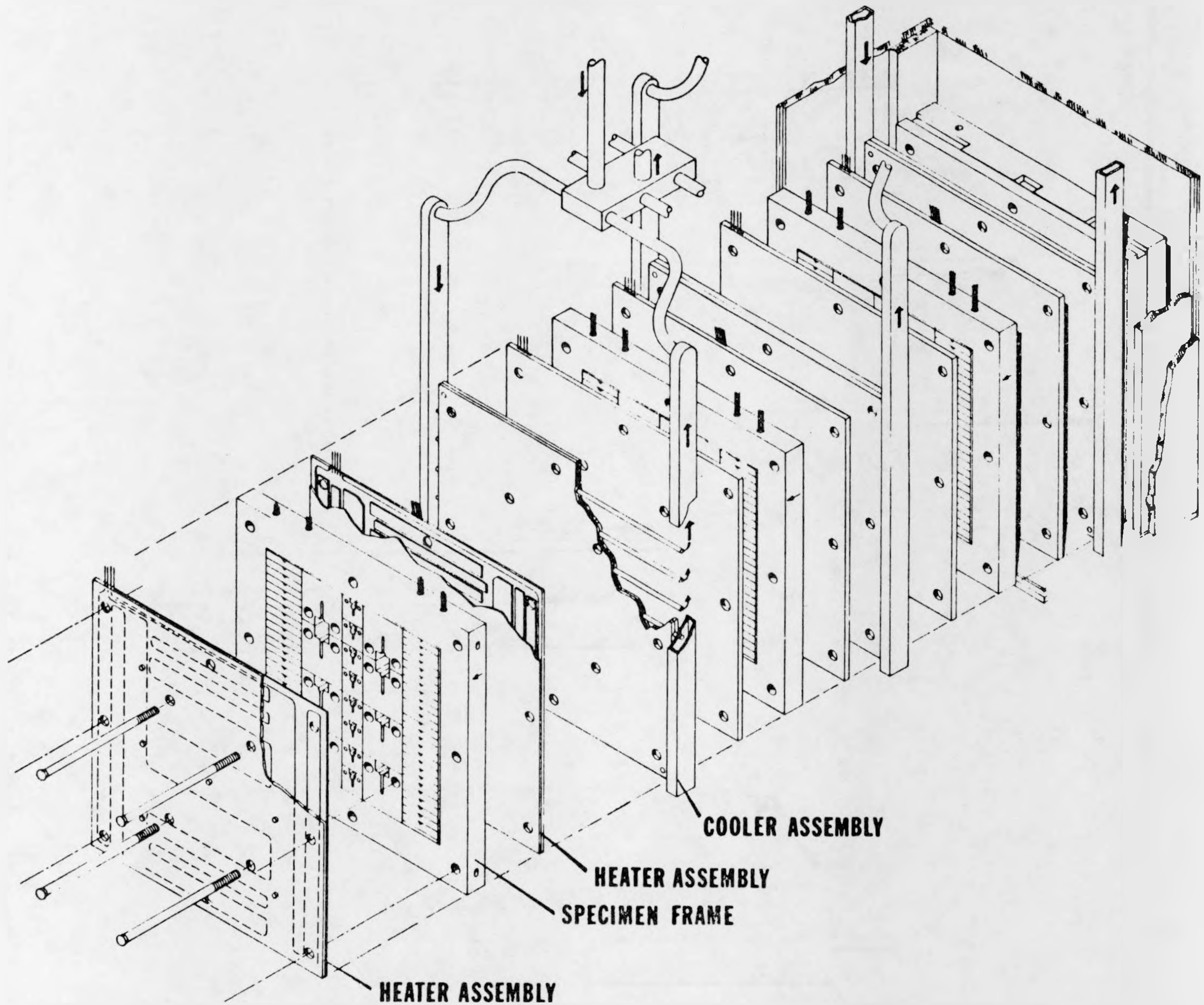


FIGURE 2.8 Exploded View of the Simulated Pressure Vessel Capsule. Neg 8010270-15.

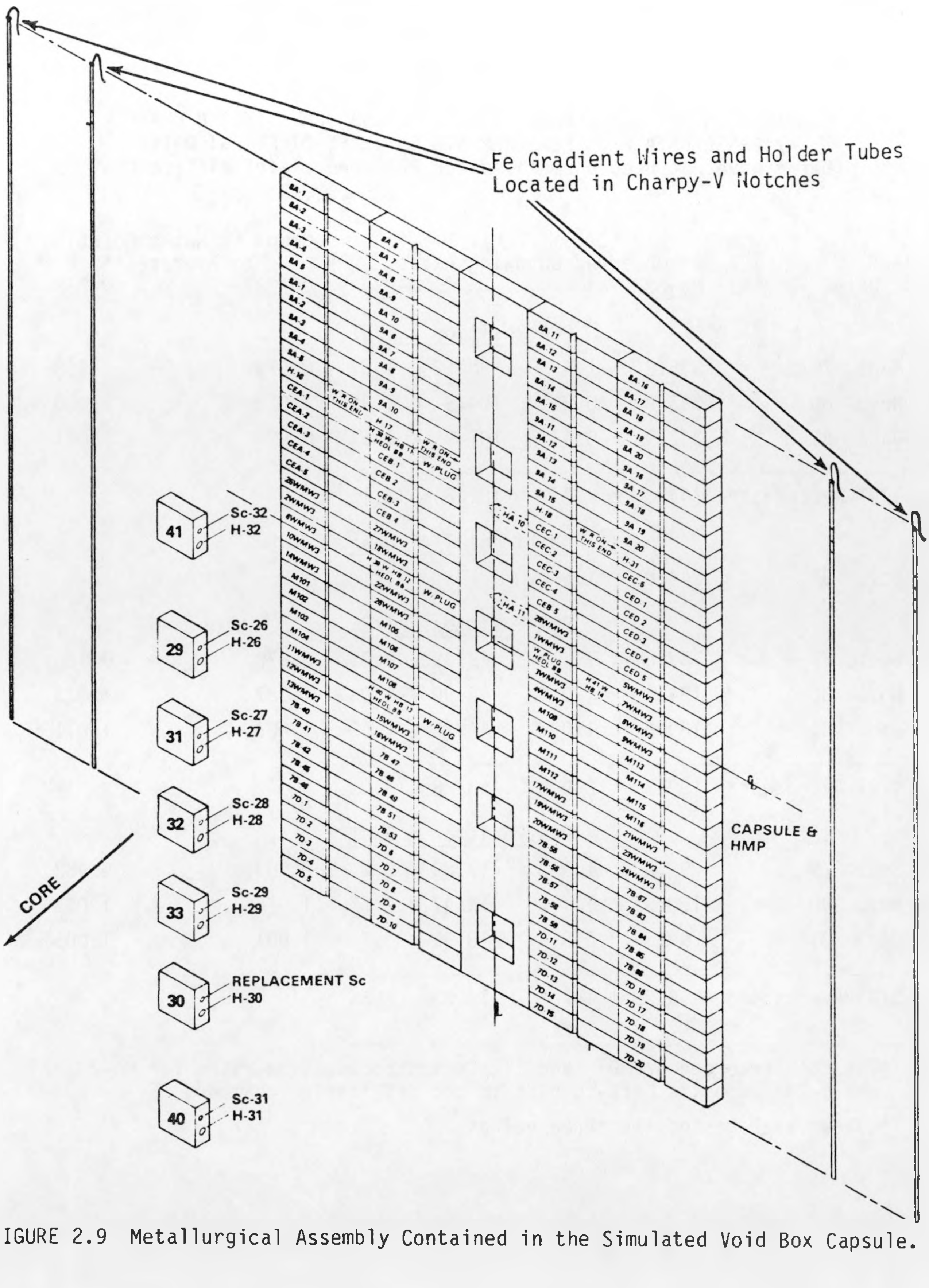


FIGURE 2.9 Metallurgical Assembly Contained in the Simulated Void Box Capsule.

TABLE 2.1

EXAMINATION OF THE REPRODUCIBILITY OF NBS DUAL FISSION CHAMBER 4/12+SSC DATA FROM MEASUREMENTS ON THREE DIFFERENT DATES
(Data* have not been normalized for PCA power level differences)

Date	Observed Counting Rates Np-237	Observed Counting Rates U-238	Counting Rates Relative to Averages** Np-237	Counting Rates Relative to Averages** U-238
		PCA Position = 1/4T		
Sept. 79	335.2 ± 1.4 %	98.52 ± 1.8%	0.974	0.968
Nov. 80	353.9 ± 0.81%	104.9 ± 1.9%	1.028	1.030
Jan. 81	342.9 ± 0.62%	101.9 ± 1.8%	0.996	1.001
1/4T Averages**	344.3 ± 2.7 %	101.8 ± 3.1%		
		Additional 1/4T Data		
Oct. 81	344.0 ± 1.6 %			
		PCA Position = 1/2T		
Sept. 79	183.8 ± 1.7 %	42.39 ± 2.0%	0.970	0.966
Nov. 80	194.4 ± 1.0 %	44.90 ± 2.0%	1.027	1.023
Jan. 81	189.8 ± 1.0 %	44.36 ± 1.5%	1.003	1.011
1/2T Averages**	189.3 ± 2.9 %	43.88 ± 3.0%		
		PCA Position = 3/4T		
Sept. 79	97.63 ± 1.9 %	17.89 ± 2.2%	0.983	0.982
Nov. 80	100.91 ± 1.1 %	18.47 ± 2.5%	1.016	1.013
Jan. 81	99.34 ± 0.8 %	18.32 ± 1.7%	1.001	1.005
3/4T Averages**	99.29 ± 1.7 %	18.23 ± 2.5%		

*The data recorded are DT- and ETZ-Corrected counting rates for Np-237 and U-238 deposits back-to-back in the NBS fission chamber.

**Linear averages of the three values.

TABLE 2.2

EXAMINATION OF THE REPRODUCIBILITY OF NORMALIZED NBS DUAL FISSION CHAMBER 4/12+SSC DATA FROM MEASUREMENTS ON FOUR DIFFERENT DATES
(Data* have been normalized to a 10.00-kW PCA power level)

Date	Observed Counting Rates Np-237	Observed Counting Rates U-238	Counting Rates Relative to Averages** Np-237	Counting Rates Relative to Averages** U-238
		PCA Position = 1/4T		
Sept. 79	338.9 ± 1.7 %	99.62 ± 2.0 %	1.004	0.998
Nov. 80	336.7 ± 1.4 %	99.81 ± 2.2 %	0.998	1.000
Jan. 81	338.8 ± 1.4 %	100.7 ± 2.2 %	1.004	1.009
Oct. 81	335.0 ± 1.9 %	99.0 ± 3.6 %	0.998	0.992
1/4T Averages**	337.4 ± 0.81%	99.78 ± 0.85%		
		PCA Position = 1/2T		
Sept. 79	185.8 ± 2.2 %	42.86 ± 2.2 %	0.998	0.994
Nov. 80	185.0 ± 1.5 %	42.72 ± 2.3 %	0.994	0.990
Jan. 81	187.5 ± 1.6 %	43.83 ± 2.0 %	1.008	1.016
1/2T Averages	186.1 ± 1.0 %	43.14 ± 1.6 %		
		PCA Position = 3/4T		
Sept. 79	98.72 ± 2.4 %	18.09 ± 2.4 %	1.011	1.009
Nov. 80	96.00 ± 1.5 %	17.57 ± 2.7 %	0.983	0.980
Jan. 81	98.16 ± 1.5 %	18.10 ± 2.1 %	1.005	1.010
3/4T Averages	97.63 ± 1.1 %	17.92 ± 2.0 %		

*The data recorded are DT- and ETZ-Corrected counting rates rates normalized to 10 kW power for Np-237 and U-238 deposits back-to-back in the NBS fission chamber.

Linear averages of three or four values.*

***Normalized void box data are still: Np-237 = 26.82 ± 1.8%
U-238 = 4.34 ± 2.1%
The 4.1% absolute power uncertainty has not yet been included.

TABLE 2.3

SUMMARY OF RUN-TO-RUN MONITOR CALIBRATIONS
FOR VARIOUS PERIODS FROM JUNE 1979* TO THE PRESENT

Period in Program History	Run-to-Run Monitor Calibration Factors** in Watts/cps of Monitor Heavier Deposit
June 1979 - January 1980	1.656 ± 0.5%***
November 1980	0.775 ± 0.8%
January 1981	0.732 ± 0.9%
October 1981	0.774 ± 0.9%
November 1981	0.759 ± 0.8%

*There are no reliable run-to-run monitor data prior to June 1979.

**Calibrations between June 1979 and January 1981 were done by A. Fabry, CEN/SCK. Subsequent to this, calibrations were done by E. D. McGarry, NBS.

***Not the same deposits as used for subsequent operations.

NOTE: In June 1981 NBS re-verified the previous CEN/SCK determination of the calibration factor relating fission chamber (7) rates at PCA core center to absolute core power. The relative CEN/SCK calibration factor was 0.01204 ± 0.00015. NBS re-verified this value as 0.0119 ± 0.0002.

TABLE 2.4

COMPARISON OF PRESENT AND EARLIER CALCULATED SATURATED ACTIVITIES IN THE PCA 12/13

REACTION/POSITION	A1	A3M	A4	A5	A6
$^{27}Al(n,\alpha)$	0.96*	1.00	0.99	0.99	1.00
$^{58}Ni(n,p)$	0.97	1.00	0.98	0.97	0.98
$^{238}U(n,f)$	--	--	0.97	0.96	0.96
$^{115}In(n,n')$	0.95	0.98	0.96	0.95	0.94
$^{237}Np(n,f)$	0.95	0.98	0.96	0.95	0.92

*Values presented are ratios of the calculated activities from the present analysis to those obtained two years ago.

TABLE 2.5

SUMMARY OF COMPARISON OF MEASURED AND CALCULATED $^{235}U(n,f)$ REACTION RATES IN THE PCA 12/13 (Fissions/Nucleaus/Core Neutron)

RESULT/POSITION	A1	A3M	A4	A5	A6
Bare Meas.	2.45-26*	8.08-28			
Bare Calc.	2.40-26	8.70-28	2.58-30	6.94-31	2.99-31
C/E	0.98	1.08			
Cd-covered Meas.	1.87-28	6.39-30			
Cd-covered Calc.**	1.71-28	6.35-30	1.30-30	6.16-31	2.80-31
C/E[b]	0.91	0.99			
Cd Ratio Meas.	131	126		1.25	1.10
Cd Ratio Calc.**	140	137	1.98	1.13	1.07
C/E[b]	1.07	1.09		0.90	0.97

*Read 2.45×10^{-26} fissions/nucleus/core neutron. MOL fission chamber results in water, HEDL SSTR results in iron. (Ref. 23.)

**Calculated assuming a cadmium cutoff of 0.414 eV. Corresponding values in water for a cutoff of 0.58 eV are about 10% less (10% more in Cd ratio). Iron values are little affected.

TABLE 2.6

PHOTOFISSION ENHANCEMENT EFFECTS IN THE PCA 12/13

RESULT/POSITION	A1	A3M	A4	A5	A6
$f_{28}(\gamma+n)/f_{28}(n)$	1.017	1.061	1.032	1.018	1.011
$f_{37}(\gamma+n)/f_{37}(n)$	1.009	1.033	1.010	1.004	1.002

TABLE 2.7

COMPARISON OF ABSOLUTE NEUTRON AND GAMMA-RAY FLUXES
IN THE PCA 12/13 FOR TWO INDEPENDENT CALCULATIONS (Particles/cm^2/Core Neutron)

RESULT/POSITION	A1	A2	A3M	A4	A5	A6
Gammas > 6.5 MeV, ORNL	1.72-6	3.57-7	2.26-7	3.14-8	7.53-9	1.89-9
Gammas > 6.5 MeV, MOL	1.26-6	4.16-7	2.50-7	4.35-8	1.04-8	1.03-8
Neutrons > 0.8 MeV, ORNL	3.95-6	4.40-7	1.43-7	5.01-8	2.42-8	1.09-8
Neutrons > 0.8 MeV, MOL	4.41-6	5.23-7	1.70-7	6.37-8	3.31-8	1.61-8
$\phi\gamma/\phi n$, ORNL	0.435	0.811	1.58	0.627	0.311	0.173
$\phi\gamma/\phi n$, MOL	0.286	0.795	1.47	0.680	0.314	0.640

3.0 ASSESSMENT, CONTROL, AND VERIFICATION OF THE PRESENT AND END-OF-LIFE CONDITION OF PRESSURE VESSELS AND THEIR SUPPORT STRUCTURES

3.1 BACKGROUND AND INTRODUCTION

A number of potential methods exist for assuring the adequacy of fracture control of reactor pressure vessel (PV) beltlines under normal and accident loads.[1-9] One of these methods, involving the use of fuel management schemes for reducing the rate of neutron damage accumulation at points of high neutron exposure, shows considerable promise.[10-12] Practices for assessing and controlling the condition of PV beltlines and their support structures follow the recommendations in the US Code of Federal Regulations 10CFR50 (App. G and H) and 10CFR21, respectively, as well as those of the ASME Boiler and Pressure Vessel Code, Sec. III and XI.[13-16]

This section of the annual report reviews the methods and regulatory requirements for fracture behavior assessment and control and the interfaces with physics-dosimetry-metallurgy. It then reviews the calculated effects of new fuel management schemes on derived exposure parameter values for a representative PWR power plant. These are followed by a review of recent and selected results of LWR-PV-SDIP interlaboratory efforts. In addition to the assessment of the condition of PV and support structure steels, this work is directed towards the verification of the effects of old and new fuel management schemes using new physics-dosimetry-metallurgy methods, procedures and data being developed, tested, verified, and recommended in a new set of ASTM Standards.[17]

3.2 METHODS AND REGULATORY REQUIREMENTS FOR FRACTURE BEHAVIOR ASSESSMENT AND CONTROL AND PHYSICS-DOSIMETRY-METALLURGY INTERFACES

Figure 3.1 is a flow sheet of the steps involved in applying the combined results of physics-dosimetry-metallurgy assessment to fracture analysis. As diagrammed in Figure 3.1 and stated by Randall in Ref. 7, "fracture analyses require material properties information, especially about neutron exposure, one that correlates with damage to the material as a function of its chemical composition, irradiation temperature, and time of exposure." Table 3.1 (data taken from Ref. 18) summarizes how property changes predicted by physics calculations and metallurgical tests are used in licensing and regulation. The table cites two measures of the state of radiation embrittlement of reactor vessels. Whenever either or both are not satisfied, continued plant operation can still be insured under the following conditions (see 10CFR50 App. G, Sec. V-C):[13] complete in-service inspection of the beltline (Reg. Guide 1.150),[18] with additional fracture toughness assessment and demonstration, followed by adequate fracture mechanics analysis[4,15,18,63,67,68] that the safety margins remain sufficient; if they are not, implementation of new fuel management schemes, a vessel anneal[1-3,64] other controls, or plant shutdown could be necessary.

Advances in steel-making and vessel fabrication technologies[2,3,9,18] have become so successful that, for recently constructed and future plants, insufficient ductile toughness or elevated ductile-to-brittle transition temperature is unlikely to be a problem during the normally planned reactor life. Some less recently constructed vessels might, however, contain weldments that will eventually become vulnerable [i.e., to a pressurized thermal shock subsequent to an overcooling accident (OCA),[18,20-22,63,67,68,90] if no remedial action were taken]. Timely reduction of vessel wall exposure by fuel management methods (low leakage cores)[10-12] can reduce or eliminate the risk of such an occurrence.

Accelerated test reactor and power reactor surveillance capsule irradiations of relevant base metal, heat-affected zone and weld specimens allow early examination of PV steel performance and contribute to the elaboration of an adequate metallurgical data base: trend curves and correlations.[2,3,6-9,62,64,65,69,70,74,75] Such irradiations generally demand excellent temperature stability and consideration of spatial and neutron flux-spectral effects. This is crucial, given the sensitivity of PV steel embrittlement to temperature, spatial, and flux-spectral parameters. These spatial and flux-spectral parameters are being extensively studied in a special series of LWR-PV-SDIP pressure vessel simulator and simulated surveillance capsule experiments, see Sections 2.2 and 2.3. Additional information on the physics-dosimetry for all of the above experiments is provided in Section 3.4. References 69 and 71 provide information on the metallurgical testing accomplished up to October 1982.

Conflicting views still exist regarding the actual temperature difference between surveillance capsules and reactor vessel walls, while heating within the reactor core internals remains ill-understood. A systematic investigation of the gamma-ray components of the PWR radiation field is in progress[23-27] and will be intensified in forth-coming interlaboratory benchmark work in VENUS,[18,24,60] in the Mol cavity fission spectrum standard field,[28] and in NESDIP,[24,29,61]; see Sections 2.2.1 and 2.2.2 for brief discussions of PCA results. Verification of the effects of fuel management methods on both the neutron and the gamma-ray components of PWR in- and ex-vessel neutron fields using these three benchmarks is an important new LWR-PV-SDIP interlaboratory effort.

Figure 3.2 is a block diagram showing the input and output flow of information for fracture analysis using the overcooling accident (OCA-I) code.[18,20-22,63,67,68] As part of this fracture analysis, the importance of selecting the correct input physics-dosimetry-metallurgy data (vessel material damage estimates) for different fuel management schemes is considered in Section 3.3.

3.3 FUEL MANAGEMENT EFFECTS AND NEUTRON EXPOSURE PARAMETERS

The benefits of low neutron leakage fuel management schemes have been studied rather extensively by the nuclear industry. At the request of NRC, HEDL has performed such a study to determine the benefits of replacing selected outer row fuel with stainless steel assemblies for reducing pressure vessel wall neutron exposures at points of high accumulated neutron damage.[10,11] Further, the NRC staff has conducted a survey of eight licensees, vendors and several foreign reactors for methods of lowering neutron exposures to pressure vessels.[12] They find that two methods in current use are 1) low neutron leakage core loading and 2) fuel assembly substitution. Based on the survey, reduction of neutron exposure of up to a factor of ∿5 appears feasible for Method (1) and up to a factor of ∿10 or more for Method (2).

As stated above, the Method (2) technique of fuel assembly substitution has been investigated by HEDL. Calculations were run for six types (A through F) of commercial generic PWRs. The reactor types were chosen primarily on the basis of the immediate availability of required information. For the purposes of this review, it will suffice to illustrate results with just the Type A PWR. The information presented is taken directly from Ref. 10.

Particular core fuel assemblies were identified as the heaviest contributors to the flux at the point on the vessel wall with the highest damage accumulation rate. A 2-D transport calculation was used to determine the benefit to be gained by replacing a few fuel assemblies by stainless steel (SS) dummies, with appropriate water fractions to account for the coolant. The core power distribution in the remaining fuel assemblies was assumed to be unchanged except for a renormalization factor that maintained the same total power output.

For the Type A PWR reactor with both accelerated and wall surveillance capsules, reactor physics calculations were made for 3 conditions: (a) full fuel, capsules in, (b) modified fuel, capsules in, and (c) full fuel, capsules out. The R and theta meshes were the same for the three cases. The core map in (x,y) geometry is shown in Figure 3.3. The calculations used as-built dimensions for a particular reactor installation. Figure 3.4 shows the Type A reactor in an (R,θ) map, which indicates the mesh detail in the DOT calculation. A comparison of Figures 3.3 and 3.4 shows that two outer fuel assemblies (a and b) in the region near $(0° < \theta < 10°)$ were replaced by SS dummies with appropriate water fractions in the modified-fuel DOT calculation. For the case of surveillance capsules out, all three capsules were removed and a normal fuel load was assumed.

Figure 3.5 compares the dpa[17,30] damage exposure dose on the front face of the pressure vessel after 32 years of full-power operation for two cases: (a) a full fuel load is assumed and (b) two fuel assemblies were replaced by SS dummies with power distribution renormalized to return to full power. The reduction in dpa damage exposure rate at the $\theta = 0°$ position is 13.6/1.0, but the peak damage accumulation is shifted to the 29° angular position. The ratio of normal to modified fuel in maximum-damage rate is 1.58/1.0.

Using dpa is an attempt to express radiation damage in a unit that can be applied to a wide variety of neutron spectra. Fluence greater than some selected energy level (e.g., E > 1.0 MeV) does not correctly account for lower energy neutrons and differences in spectral shapes in general. The significance of this consideration to the neutron exposure of pressure vessels throughout their thickness is indicated in Figure 3.6. In this figure we have taken a radial sweep from the core out through a surveillance capsule at the $\theta = 35°$ angular position. We have calculated dpa/ϕt (E > 1.05 MeV), normalized to unity at the capsule center. As can be seen from Figure 3.6, the dpa/ϕt ratio at the 1/4 T position is about 20% higher than the ratio at the surveillance capsule position and the dpa/ϕt ratio varies by a factor of 2.23 going from the PV front to the rear. Therefore, if ϕt (E > 1.0 MeV) information is used with surveillance capsule mechanical properties data to develop in-vessel material property change trend curves, the conclusions drawn from such information will be nonconservative. Also exposures will be nonconservative by a factor of two if the trend curve is used with ϕt (E > 1.0 MeV) exposure information for positions near the PV rear wall. More information on this subject is provided in Table 3 of Ref. 17 for PWR, BWR and Test Reactor neutron fields.

For the Type A Reactor accelerated and wall surveillance capsules, each capsule was modeled as a 15-region rectangle (3 theta regions x 5 radial regions). The center lines of the capsules are located at 3°, 35° and 45°. The capsule perturbation effect can be seen in Figures 3.7 and 3.8, in which radial traverse values of dpa are plotted at the 3° and 35° angular positions; capsule in is compared with capsule out for the 32-year full-power dpa exposure. The capsule's presence causes an increase in neutron exposure, measured either in dpa or in fluence (E > 1.0 MeV) units. At the capsule center, its presence causes an increase of about 24% in the dpa exposure value or an increase of about 23% for the fluence (E > 1.05 MeV) for the wall capsule located at an angular position of 3°. For the accelerated capsule located on the core side of the thermal shield, the similar increases are about 27% for the dpa exposure and about 24% for the fluence (E > 1.05 MeV). These types of calculated perturbation effects have been verified for the Westinghouse and Combustion Engineering simulated dosimetry measurement facility (SDMF) perturbation and the first ORR-SDMF RM sensor certification test (Figures 3.16 and 3.17). For this type of power plant and surveillance capsule configuration, transport code solutions obtained without explicit capsule modeling will require corrections of the magnitude indicated above, when the transport solution is used directly to provide a "lead factor."

For surface flaws, the relative importance of using the dpa exposure parameter in estimating present and EOL shifts in ΔRT_{NDT} is indicated by the solid and dashed curves shown in Figure 3.9. For an assumed set of PWR overcooling accident parameters, different results are obtained for the maximum pressure to permit crack arrest by using dpa instead of fluence (E > 1.0 MeV) as the exposure parameter to determine the ΔRT_{NDT} shift as a function of distance in the pressure vessel wall. It is seen that the use of dpa would reduce the allowable shift in ΔRT_{NDT} by about 2% at high pressures (>1000 psi) and about 7% at low pressures ($\sim$250 psi) for the conditions assumed for this particular transient. The significance of such changes will be dependent on the sceening criteria selected by NRC (and other licensing and regulatory bodies) for the allowable value of $RT_{NDT} = RT_{NDT} + \Delta RT_{NDT}$, see Table 3.1

Based on the above and the information presented in Section 2.4 and Reference 24, it is apparent that controlling variables can change, but that those associated with the determination of the spatial (lead factor) and exposure time (trend curve) extrapolations of power reactor surveillance capsule (and research reactor test) data are extremely important. Further, the relative importance of any one variable will be dependent on a number of factors, i.e., 1) the reliability and applicability of available physics-dosimetry-metallurgy data bases for individual power plants, 2) the plant design, safety, and operating conditions, and 3) the relative importance of specific licensing and regulatory issues and criteria. More detailed information and results on the effects of using dpa and different trend curve power law dependences as input to the OCA-I code will be found in References 20-22 and 63. Also, a copy of a Research Information Letter (RIL) prepared by HEDL for NRC on the "DPA Exposure Unit" is reproduced in the Appendix, Section 5.0, for reference and comment purposes.

From information provided in Figures 3.1 through 3.9 and in Table 3.1, the significance to the nuclear industry of the determination and verification of the effects of using old and new fuel management schemes and different exposure parameters on the assessment and control of the condition of PV and support structure steels is readily apparent. That is, timely reduction of vessel wall exposure by fuel management methods (low leakage cores) provides a practical and perhaps relatively inexpensive approach to reducing or eliminating the risk of fracture associated with pressurized thermal shock. It should be noted that low leakage core designs were initially proposed for economic reasons (increased fuel burnup), and their effect on ex-core component neutron exposure has since been recognized. The following Section 3.4 deals with the results and status of work on verifying the effects of using different fuel management schemes and exposure parameters by application of the new physics-dosimetry-metallurgy methods, procedures and data being developed, tested, verified, and recommended in the set of 21 ASTM LWR standards (Figures 3.10 and 3.11).

3.4 ASTM STANDARDS, RECOMMENDED PROCEDURES AND DATA, AND RESEARCH AND POWER REACTOR VERIFICATION STUDIES

3.4.1 ASTM-Recommended Methodology

To verify the accuracy of predictions of the present and end-of-life (EOL) condition of PV and support structure steels requires the application of newly developed, tested, verified, and recommended ASTM methods, procedures and data. To accomplish this, the present work strategy depends on international inter-laboratory participation using research and development, standards, and commercial technology applications and assessment.[1-94] The important "educational" value of this work (i.e., transfer of technology) keeps the research and development, standards, and commercial developments "in tune" with the evolution of licensing and regulatory views and the needs of nuclear power plant vendors, architect/ engineers, utilities, and service laboratories.

Table 3.2, taken from the new ASTM E853-81, "Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results"[39] (see Figures 3.10 and 3.11 and Section 2.1.1), summarizes the methodology to be used in the analysis and interpretation of neutron exposure data obtained from LWR surveillance programs and, based on the results of that analysis, establishes a formalism to be used to evaluate the present and future conditions of pressure vessel and support structure steels. The interrelationship of some of the new physics-dosimetry standards for the determination of exposure values is represented by the block diagram in Figure 3.12.

As stated in Table 3.2, Step 4, it is necessary to establish a reactor physics computational method applicable to the surveillance program of a particular plant. Table 3.3 shows the established computational methods, nuclear data, and exposure values most often used and reported for a number of U.S. laboratories and vendors. Similar information for other laboratories is provided in Refs. 23 and 31 through 38.

References 23 and 40 provide state-of-the-art information about the accuracy (5 to 30%, 1σ) of exposure parameter values that can currently be obtained by the use of new neutron spectrum adjustment codes, ASTM E706(IIA) Standard (the central box of Figure 3.12). Table 3.4 and Figures 3.13 through 3.15, taken from Refs. 24 and 40, provide summary results on the obtainable accuracies for exposure parameter values for PWR surveillance programs as of October 1982. Some of the reevaluated fluence ($E > 1.0$ MeV) values with the FERRET-SAND adjustment code assigned uncertainties were combined with selected surveillance capsule measured reference steel (A302B and A533B correlation monitor material) Charpy-V ΔRT_{NDT} results and compared with Reg. Guide 1.99 trend curves[8] (see Figure 3.15). For the Point Beach 2 data, the reevaluated results are shifted to the right and are, therefore, conservatively bounded by the Reg. Guide curve (0.14% Cu/0.012% P). This is generally the case for all the other reevaluated exposure parameter and ΔRT_{NDT} data shown in the figure.

Having established a reactor physics computational program, input information is needed on the cycle-to-cycle core power distribution (Steps 5 and 6 of Table 3.2), which is dependent on both old and new fuel management schemes used by different utilities. This in turn will have an important effect on determining and verifying surveillance capsule perturbations, photoreaction and other corrections (and effects) for individual radiometric (RM), solid state track recorder (SSTR), helium accumulation fluence monitor (HAFM), and damage monitor (DM) sensors (Steps 7 and 8).[17,18,23-25,28,29,31-38,40-43,46-54,66,72,73] The individual ASTM standards for the application and analysis of these four methods are identified in Figure 3.12. Also shown is the ASTM standard for temperature monitors (TM), which is needed for determination and verification of the irradiation exposure temperature of DM sensors and surveillance capsule metallurgical specimens.

The final steps (8 through 11 of Table 3.2) involve benchmark validating the analytical methods, establishing methods for relating and verifying the accuracy of physics, dosimetry, metallurgy and temperature data from surveillance programs. In Table 3.5, taken from Ref. 24, benchmarks are identified along with the development time frame, participants, and their intended purpose and use. The discussion that follows in Sections 3.4.2 and 3.4.3 will concentrate on the presentation of summary information on the status of LWR-PV-SDIP interlaboratory program work for "Surveillance Capsule" and "Generic Test and Power Reactor" benchmarks, up to October 1982.

3.4.2 Surveillance Capsule Benchmark Perturbation and Radiometric Reaction Rate Studies

To provide experimental verification of reactor physics predictions, a perturbation experiment was performed in the Simulated Dosimetry Measurement Facility (SDMF) at the Oak Ridge Reactor (ORR) Pool Side Facility (PSF).[24,41] This second ORR-SDMF test (the startup test was the first one) was used to simulate the perturbation effects for Westinghouse and Combustion Engineering surveillance capsule designs. (This test also served as the first US ORR-SDMF RM sensor certification test while the ORR-PSF first simulated surveillance capsule [SSC-1] metallurgical irradiation served as the second US RM sensor certification test.) The experimental mockup design is shown in Figure 3.16, taken from References 24 and 41.

Based on the measured reaction rates and on the measured azimuthal flux distribution and with the aid of spectrum-averaged cross sections calculated according to the principles shown in Ref. 41, values of flux (E > 1 MeV) were derived in the perturbed and unperturbed thermal shield back (TSB) and pressure vessel front (PVF) positions. The results are shown in Figure 3.17, taken from Ref. 41. The neutron flux values of the ^{237}Np(n,f), ^{238}U(n,f), ^{58}Ni(n,p), ^{54}Fe(n,p) and ^{46}Ti(n,p) detectors agree within 5%. The spectrum-averaged ^{63}Cu(n,α) cross sections are somewhat overestimated in the TSB position, while the spectrum-averaged ^{93}Nb(n,n') cross sections are somewhat underestimated in the PVF position. The agreement between experiment and calculation is excellent, better than 5% for all ratios except ^{93}Nb(n,n'). The somewhat different behavior of the calculated ^{93}Nb(n,n') data can only be explained by an incorrect shape for the ^{93}Nb(n,n') cross-section curve. These results verify that current reactor physics procedures and data can be used to provide reliable estimates of perturbation effects for Westinghouse and Combustion Engineering type surveillance capsules, see also Sections 2.2.2 and 2.3.2.

A third interlaboratory ORR-SDMF test irradiation was completed in September 1982 for B&W surveillance capsule designs. (This test will also serve as the third US RM sensor certification test.) Figure 3.18 is a picture of the as-built experimental mockup for two B&W surveillance capsules. A rather extensive set of RM, SSTR, HAFM and DM sensors were used for this perturbation experiment. Results of the calculated-to-measured values of sensor reaction rates and the derived values of flux levels above 1.0 MeV will be available in mid-1983. RM sensor certification test results will be available somewhat later, depending on vendor and service laboratory analysis and reporting schedules.

Tables 3.6, 3.7 and 3.8, taken from Refs. 42 and 43, provide summary RM sensor LWR-PV-SDIP interlaboratory comparison results for the first and second US ORR-SDMF RM sensor certification tests. It is noted that the ORR-PSF-SDMF startup test served as the first European RM sensor certification test. For the Tables 3.6 and 3.7 results, HEDL served as the reference counting laboratory for the comparisons of four US vendors and two US service laboratories. For the Table 3.8 results, CEN/SCK was the reference counting laboratory for four European laboratories. For the Tables 3.6 and 3.7 comparisons, the preliminary results (shown only in Ref. 42) were distributed over a range of relative values as large as 60%. Had results from a single laboratory been used to derive surveillance capsule fluence values, which are often based on only one or two reactions, a bias of 40% or more could easily have been introduced. Following discussions of the preliminary analysis results and identification of problem areas, the biases were generally reduced to below 15%. The final results and conclusions for the Table 3.8 European comparisons are similar to those for the final US comparisons.

While agreement among the majority of the laboratories participating in the above interlaboratory reaction rate comparisons is generally good, improvement is still required in order to routinely meet surveillance program requirement goals of 5 to 10% (1σ) for measured reactions and reaction rates. Without such sensor set input accuracy for adjustment codes, in- and ex-vessel surveillance position exposure parameter values cannot be derived with the necessary 10 to 30% (1σ) accuracy. Other uncertainties will be added to surveillance position point-wise data in extrapolating physics-dosimetry-metallurgy results to other reactor beltline locations of interest. It is necessary, therefore, to keep surveillance capsule exposure parameter value acccuracies in the 10 to 20% (1σ) range.[24,40] The process of extrapolating from a surveillance capsule position to the pressure vessel inner wall and then to different axial and azimuthal positions of interest can add another 10 to 30% (1σ) uncertainty to the final exposure parameter values used for fracture analysis studies.

The results obtained from these three ORR-SDMF tests along with subsequent corrections indicate that a critical review of both analytical counting and calculational techniques must be conducted on a periodic basis by all surveillance program service laboratories. In addition, it is necessary that each involved laboratory review and utilize, where possible, the appropriate ASTM Standard Practices, Methods and Guides (see Figures 3.10 through 3.12). This will be necessary to maintain system calibrations, quality control and appropriate documentation, and to properly utilize existing benchmark facilities for the long-term on-going verification and certification of the accuracy of service laboratory capabilities.

3.4.3 Generic Test and Power Reactor Benchmark Studies

3.4.3.1 Power Reactor Benchmarks

3.4.3.1.1 PWR and BWR Power Reactor Benchmarks and RM, SSTR, HAFM, and DM Dosimetry

To provide experimental verification of operating PWR and BWR reactor physics-dosimetry predictions, a series of generic power reactor benchmark studies has been undertaken by the utilities, reactor vendors, government agencies and laboratories of the participating countries.[18,23,24,29,31-38,40,66,72,73] Table 3.9 lists the power reactors presently being used for these studies by LWR-PV-SDIP participants. Listed across the top are the names of the power plants, reactor type, reactor supplier (vendor), reactor operator (utility), and ex-vessel cavity (C) or in-vessel (V) surveillance positions available for dosimetry measurements.

The first three columns list the energy response ranges, type of dosimeters (RM, SSTR, HAFM, DM), and dosimetry reactions of current interest. In the body of the table, blank columns indicate that any additional in- or ex-vessel monitoring has not been attempted or is either not possible or feasible at this time. "Y" in a block indicates "yes" (this type of dosimetry has been used); "P" indicates that it is planned; "N" indicates it was not desired or cannot be used. Any of the forenamed letters (Y,P,N) within parentheses suggest some doubt. For example: (Y) in the $^{238}U(n,f)$ ^{137}Cs column of a cavity irradiation suggests that there may not be sufficient fluence in a single reactor cycle to produce a reliable measurement of ^{137}Cs for a radiometric dosimeter. Consideration of the use of other RM, SSTR, HAFM or DM monitors might, therefore, be in order.

The status and results of a number of EPRI-NRC supported power reactor benchmark studies for the verification of reactor physics predictions are discussed in Refs. 32 through 38, and the reader is referred to these references for more detailed information. References 18, 66, 72, and 73 provide information on CEN/SCK-NRC-EPRI benchmark studies for BR3. Here, we will only consider some of the planning and/or results for the ANO-1, H. B. Robinson, Maine Yankee, and the Crystal River (or Davis-Besse) studies, Table 3.9. The discussion of ANO-1 results concentrates on sensor set selection and exposure parameter response ranges. These results are presented in Section 3.4.3.1.3. Only the H. B. Robinson, Maine Yankee, and Crystal River (or Davis-Besse) benchmark tests have been designed to provide direct experimental verification of the accuracy of reactor physics-dosimetry predictions for new low leakage core fuel management schemes. The status of LWR-PV-SDIP interlaboratory program work for these three benchmarks is discussed in more detail in Section 3.4.3.3. The selection of RM, SSTR, HAFM, and DM sensors for these three power plants is indicated in Table 3.9, for both the surveillance capsule and cavity positions. Experimental results for these three tests will not be available before another one or two years, depending on the power plants' operating schedules.

3.4.3.1.2 Dosimetry Sensor Selection and ASTM Standards

The determination and selection of an appropriate set of RM, SSTR, HAFM, and DM sensors is an important and critical step for in-and ex-vessel LWR power plant surveillance programs. This selection is affected by many questions and issues: 1) Are the costs reasonable and is the quality assurance (QA) information on sensor purity and target atom content per unit weight well documented and verified? 2) Are temperature stability and retention of reaction products of interest over many years (up to 30 to 50) well enough known? 3) For RM sensors, are reaction product half-lifes long enough? 4) Are handling, shipping and licensing requirements for fissile and radioactive sensor materials manageable, known and in place? 5) Are the accuracy of nuclear data and neutron energy response adequately known? i.e., is the sensor set selected able to measure the neutrons that cause damage in PV and support structure steels and do the measured results properly correlate with the observed metallurgical property changes? and 6) Are standardized procedures, data, and recommended documentation and reporting requirements available to the utilities, vendors, and service laboratories performing the counting and analysis of individual sensors? That is, can they properly maintain the required RM, SSTR, HAFM, and DM equipment calibrations to achieve 5 to 10% (1σ) accuracy on individual sensor measured reactions and reaction rates over the long-term (30 to 50 years) and provide the necessary confirmatory documentation; i.e., as employees change jobs or retire and new people enter the field.

To provide the answers to the above series of questions, in particular, the counting, analysis, and reporting of sensor results, four new ASTM standard methods are in the preparation, testing, and verification process. They are identified in Figures 3.10 through 3.12 as the RM, SSTR, HAFM, and DM ASTM standards IIIA, B, C, and D, respectively. The first version of the new SSTR standard has been accept by ASTM and given the designation ASTM E854-81 and is available in the 1982 Annual Book of Standards.[44] The current schedule for the preparation, verification, and revision of these four standards is shown in Figure 3.11, and is discussed in Section 2.1.1.

3.4.3.1.3 Arkansas Power and Light Nuclear One-1 (ANO-1)

3.4.3.1.3.1 Neutron Spectra at Cavity and Surveillance Positions

The reader is referred to Refs. 35 and 36 for the most recent discussion of ANO-1 cavity physics-dosimetry studies and results. Figure 3.19, taken from Ref. 32, shows the results of combining reactor physics predictions of the neutron spectrum for the reactor cavity and PV wall 1/4 T location with the iron dpa cross section. The cavity and 1/4 T spectra are shown with a ^{235}U fission spectrum slope for comparison. It is noted that:

- Between 1 and 3 MeV, a steeper slope versus energy in the cavity spectrum departs from the 1/4 T spectrum by a maximum of $\sim$30%.
- The hardest spectrum components are parallel above 3 MeV, an energy region which encompasses $\sim$25% of the 1/4 T dpa response.

- Below 1 MeV the iron resonance structure within the vessel is preserved in the cavity spectrum.

It may be noted that spectra at positions deeper within the vessel (not shown here) resemble even more closely the cavity spectrum. The spectrum at the surveillance position is shown in Figure 3.20. It is noted that:

- Above 1 MeV departures from the 1/4 T spectrum are as large as 20%.

- Below 1 MeV the iron resonance structure is barely apparent as the spectrum relaxes into a smooth 1/E-distribution.

In Figure 3.21 a comparison of the U-Mo., 1-D calculation, with a more detailed calculation of the PCA (12/13 configuration) provided by ORNL shows the degree to which the less sophisticated 1-D calculation is adequate as a calculational base for this review.

3.4.3.1.3.2 Concrete Albedo and Spatial Flux Distribution

Calculation results for ANO-1, performed with and without the concrete shield, are shown in Figure 3.22 for the cavity position. The spectrum differences are not large. These 2-D calculations (the only exception to the use of 1-D results in Ref. 32) indicate that about one third of the flux above 1 MeV at the detector position in the cavity is from the concrete; for a detector position near the outer surface of the pressure vessel about 12% of ϕ(>1 MeV) is from the concrete.

Flux traverses for two spectrum components, ϕ(>67 keV) and ϕ(>1 MeV) are, shown in Figure 3.23. The position of the detector capsule for the ANO-1 measurements was at a radius of 324 cm, closer to the concrete (at 345 cm) than to the vessel. The flux falls rapidly from core edge out thru the vessel and then becomes almost constant in the cavity.

The difficulty of extrapolating with confidence from the surveillance position (outer edge of thermal shield) to the in-vessel positions is well illustrated. The possibility of extrapolating back from the cavity might appear preferable since the extrapolation to 1/4 T is smaller from the cavity (much smaller to 1/2 T) than it is from the surveillance position, and it would be possible in principle to establish in-situ the position of a cavity dosimetry capsule relative to the outer surface of the vessel. The mild flux gradient in the cavity is a further aid to meeting the familiar problems of establishing local detector positions and flux perturbations. Whatever the relative merits of cavity and surveillance positions, it seems clear that neutron flux measurements on both sides of the vessel will provide essential verification of neutron transport calculations.

In summary, calculations of core leakage neutrons diffusing through surrounding water and steel and out into the pressure vessel cavity show that the neutron flux in the cavity, in conjunction with that of the surveillance position, are similar to and bracket well, the neutron flux at the

1/4 T position within the vessel, in terms both of spectrum and flux intensity. They are in important respects complementary. Iron resonance structure within the vessel is preserved in the cavity and the gradients are mild while at the surveillance position the spectrum above 1 MeV matches the 1/4 T spectrum better than in the cavity but the iron resonance structure is undeveloped and gradients are severe. This complementarity could prove useful for establishing confident measurements in support of neutron exposure lead-factors which presently are derived from calculations which often are not plant specific.

3.4.3.1.3.3 <u>RM Sensor Response Charts for Cavity and Surveillance Positions</u>

A chart of energy response ranges for representative RM threshold detectors (plus one low-energy capture detector) in the ANO-1 cavity spectrum is shown in Figure 3.24. The cavity spectrum and dpa response ranges are included in order to help evaluate spectrum coverage features of the detector set. It is noted that:

- The spectrum itself is not very relevant for neutron irradiation damage assessment and not much better incidently as a spectrum truncated to flux above 1 MeV. A spectrum weighting function related to the material property change under surveillance is necessary. The iron dpa cross section serves this purpose for steel embrittlement induced by prolonged exposure to neutrons.

- Detector set coverage of the iron dpa response range is >90%, without Np it would drop to 25%, so Np is a vital detector.

- Low energy detectors--$^{59}Co(n,\gamma)$ and $^{58}Fe(n,\gamma)$ are typical examples--are not as vital since their response barely reaches the dpa response range, but they do provide valuable data for interpolation of results in the 0.01- to 1-MeV energy range and for defining thermal and epithermal neutron flux and fluence values.

- More than about four threshold detectors leads to redundant spectrum coverage in view of the extent of detector response ranges. The ^{58}Ni and ^{54}Fe detectors, for example, are redundant and are employed as a result of experimental conservatism.

In regard to the effects of lower energy neutrons and the third item above, Figure 3.25, taken from Ref. 56, shows some recent Saclay results on the influence of thermal neutrons on the embrittlement of A537 steel irradiated at 60°C in the El.3 heavy water reactor with a thermal to fast (E > 1.0 MeV) neutron flux level ratio of ∿2000. These new results suggest that thermal neutrons will not contribute significantly to pressure vessel or support structure steel neutron damage for low temperature irradiations. Other results, however, suggest that there may be an effect for higher (∿288°C) temperature irradiations. More information on the expected effect of low energy neutrons is provided in Reference 65, Section 2.4.1, and the Appendix, Section 5.0.

The main focus of dosimetry measurement is still the survillance position. A detector response range chart for representative RM detectors is given in Figure 3.26. It is noted that:

- Detector set coverage and distinctiveness of detector responses is similar to that for the cavity spectrum, even though the dpa response range is somewhat higher.

- The Np detector is pivotal: with it detector set coverage of dpa is 90%, similar to the cavity; without it, coverage is 50%, much less but still substantially more than in the cavity.

With the above in mind, the bar charts of iron dpa-response (i.e., the atom displacement cross section) at six locations beginning with the surveillance position and ending with the detector position in the cavity are presented in Figure 3.27. The pattern of these response ranges provides additional perspectives on the cavity as a dosimetry measurement site:

- Lower bounds are all within $\pm$100 keV; median energies are more spread about.

- Less than 20% of the cavity response range is outside that of the 1/4 T position; for the 3/4 T position less than 10% is outside.

- The strong shift of the response range at the vessel inner surface may be significant because of concerns for thermal stress and possible surface cracks.

- The amount of damage exposure below 1 MeV, which varies within the vessel from 20% to 70%, calls for the use of iron dpa in addition to ϕ(>1 MeV) as a damage exposure parameter.

3.4.3.2 Test Reactor Benchmarks and RM, SSTR, HAFM, and DM ASTM Standards Verification Studies

Figures 3.28 and 3.29 are photographs of advanced RM, SSTR, HAFM, and DM in- and ex-vessel surveillance capsule dosimetry currently being evaluated in the ORR-PSF 2-year pressure vessel simulator (PVS-high power PV and support structure steel) and accelerated simulated surveillance capsule (SSC) physics-dosimetry-metallurgy tests. The irradiation phase of the PVS test was completed on June 22, 1982. Prior to this date, two accelerated simulated surveillance capsule (SSC-1 and SSC-2) irradiations were completed. The expected neutron exposures (fluence E > 1.0 MeV) for the PVS test were $\sim 4 \times 10^{19}$, $\sim 2 \times 10^{19}$, $\sim 1 \times 10^{19}$ and $\sim 5 \times 10^{17}$ n/cm^2 for the PV front, 1/4 T, 1/2 T, and cavity rear face positions, respectively. The corresponding exposures for the SSC-1 and SSC-2 surveillance position were $\sim 2 \times 10^{19}$ and $\sim 4 \times 10^{19}$ n/cm^2, respectively. Currently available metallurgical testing results for SSC-1 and SSC-2 steel specimens are discussed in References 69 and 71.

Before the PVS and SSC tests, an initial SDMF-PVS-SSC dosimetry start up test was performed.[4] Some of the RM results of this test (identified as the first SDMF test) were considered in Section 3.4.2. In support of these high power tests, are a series of low power physics-dosimetry tests in the ORNL Pool Critical Assembly (PCA).[23,24,76,78,80] In addition to the PCA-PSF-PVS and -SSC tests, advanced RM dosimetry has been evaluated in the Bulk Shielding Reactor (BSR) heavy section steel technology (HSST) dosimetry mock-up tests.[46,83,85,88] Figure 3.30 shows the experimental configurations for the BSR-HSST and PCA-PSF dosimetry mockup tests. Figure 3.31 demonstrates the applicability of the HSST results to the planned interlaboratory evaluations for the SSC and PVS 1/4 T, and 1/2 T positions.

Table 3.10 defines the individual sensor mathematical formalism and analytical methods that are being used for these PCA-PSF-HSST physics-dosimetry evaluation studies.[46] Some of the earlier results are given in Tables 3.11 and 3.12 and Figure 3.32. Information on the experimental and calculated radial fission flux attenuation by steel in typical LWR pressure vessel environments, including the Table 3.9 BR3 Belgian power plant,[18,66,72,73] are provided in Figure 3.33. These results have been compared, Figure 3.34, with some earlier CEN/SCK PCA active neutron spectrometry results based on the $^{6}Li(n,\alpha)$ technique.[23] The results of these studies are consistent with those presented in Section 3.4.3.1.3 for ANO-1.

The information provided in Figures 3.30 through 3.34 and Tables 3.10 through 3.12 of this section, Tables 3.6 through 3.8 of Section 3.4.2, and the results of other studies[31] demonstrates that the technology is well advanced for the application of RM sensors for the verification of the physics-dosimetry-metallurgy results of LWR PV and support structure surveillance programs. It has been further shown that RM results can be well correlated with steel, graphite, tungsten, silicon, and sapphire neutron damage measurements.[47-50] Thus, these individual DM sensors can be very effectively used as neutron as well as damage monitors to complement RM, SSTR, and HAFM sensors. In this regard, steel and sapphire DM sensors should be used on a routine basis for LWR power plant physics-dosimetry-metallurgy surveillance programs. What remains to be done is to complete and document the present test reactor benchmark field confirmatory studies and make sure the appropriate interfaces and procedures and data are included and recommended in the new ASTM RM, SSTR, HAFM, and DM standards.

From the information provided in Table 3.9 and, elsewhere,[23,24] it is seen that SSTRs complement the RM sensors in providing another and more sensitive technique for obtaining fission reaction results for both in- and ex-vessel measurements. The reader is referred to the new ASTM E854-81 Standard for detailed information on the use of SSTR sensors.[44] Additional and new information on the current procedures, data and benchmark fields being used to apply and test these monitors for LWR power plant surveillance are given elsewhere and need not be reviewed here.[23,24,34,38,45,51,52,55] Again, as with DM sensors, SSTR sensors should be used on a routine basis for the verification of LWR surveillance program results. As new PWR and BWR physics-dosimetry results are obtained with these sensors, evaluated, and applied, appropriate procedural and reference data revisions in the current ASTM E854-81 Standard can be made to both maintain and improve the overall accuracy of this technique.

With reference to Table 3.9, the HAFM sensors complement the RM, SSTR, and DM sensors, again for both in- and ex-vessel measurements. Table 3.13 shows the typical 90% energy response ranges for a set of eleven HAFM sensors for a fission neutron spectrum. Certainly, HAFM sensors are not as advanced as RM and SSTR in their development and testing for LWR surveillance program applications. However, the technology appears to be readily at hand[24,53,54] and appropriate sets of these detectors should be used on a routine basis for LWR surveillance program measurements. In this regard, the S(n,He) and Ca(n,He) reactions could prove to be just as beneficial as the ^{237}Np or ^{93}Nb reactions for measuring neutrons down to the 0.5 MeV energy range. If this were shown to be the case, their routine use would eliminate many of the problems presently associated with the handling, licensing, and application of fissile RM monitors.

Plans are now being made for a fourth interlaboratory SDMF test in early 1983 that would be used to provide additional experimental verification of how well RM, SSTR, HAFM, and DM sensor results can be correlated with the calculated iron dpa gradient from a surveillance capsule to and through the pressure vessel wall and to support structures. This is considered important because of the information presented in Sections 3.2 and 3.3 and the fact that the previous series of PCA-PSF-PVS experiments were designed to obtain combined physics-dosimetry-metallurgy pressure vessel wall (550°F) information only up to the 1/2 T position.[55] The fourth SDMF test would be run at ambient temperature (∿95°F) and would use an appropriate combination of RM, SSTR, HAFM, and DM sensors to accomplish the necessary physics-dosimetry verification. As a side benefit, the extreme spectral differences from the front to the back of the PVS for this test would allow better evaluation and allow for the adjustment of the energy dependent cross sections for specific RM, HAFM, and DM sensors.

3.4.3.3 LWR-PV-SDIP Verification Studies for Old and New Fuel Management Schemes and Regulatory Demands

Figure 3.12 showed the interrelationship of the new ASTM standard methods for the application and analysis of radiometric (RM), solid state track recorder (SSTR), helium accumulation fluence monitor (HAFM), and damage monitors (DM) to the determination and verification of neutron exposure parameter values. Using these new ASTM recommended procedures and data, the results of LWR-PV-SDIP verification studies were summarized by the information presented in Section 2.0 and Figures 3.13 and 3.14 and Table 3.4 for the period up to October 1982.

As previously stated, new H. B. Robinson, Maine Yankee and Crystal River (or Davis-Besse) benchmark tests have been designed to provide direct experimental verification of the accuracy of reactor physics-dosimetry predictions for new low leakage core fuel management schemes. Table 3.9 listed the power reactors being used by LWR-PV-SDIP participants to benchmark physics-dosimetry procedures and data for pressure vessel and support structure surveillance for both old and new fuel management schemes.

The planning (P), selection [Y for yes, N for not desired or cannot be used, and any of the forenamed letters (P, Y, N) within parantheses suggest some doubt], and fabrication of RM, SSTR, HAFM, and DM sensor sets for H. B. Robinson and Maine Yankee are completed. The placement of the sensor sets for H. B. Robinson has been completed and the one (or more) cycle, low leakage core, irradiation has started. Figures 3.35 through 3.38 show photographs of as-built dosimetry and the locations for placement in the in-vessel physics-dosimetry surveillance capsule and the reactor cavity. This placement was completed in June 1982.

For Maine Yankee, the placement and start of irradiations has yet to be accomplished. Figures 3.39 through 3.42 show photographs of the as-built dosimetry for a replacement physics-dosimetry-metallurgy surveillance capsule and the reactor cavity. The new surveillance capsule, which is planned for irradiation in a previously removed surveillance capsule wall location, will be held in reserve for future use, pending the establishment of an equilibrium low leakage core burnup distribution. The one or more cycle irradiation for the cavity RM, SSTR, HAFM and DM sensor sets is expected to start in late CY 1982.

Planning for the Crystal River (or Davis-Besse) benchmark studies has been initiated and actual selection, fabrication, and placement of sensors and metallurgical specimens could be accomplished in early 1983.

In support of these old and new type fuel management verification studies are a series of planned benchmark studies in the Mol Belgium VENUS, Figure 3.43, and United Kingdom NESDIP, Figure 3.44, benchmark fields.[18,29,60,61] Related to these benchmark studies, two considerations will be briefly discussed: core management benchmarking plans and lead factor assessment.

The lead factor between surveillance capsule and vessel wall is a complex parameter to determine at the required goal accuracy of 10 to 20% (1σ). If combined with a surveillance capsule accuracy of, say, 15%, this translates to a corresponding PV weld fluence accuracy of 18 to 25% (1σ). It can be conceptually separated into four parts or factors:

$$\text{Radial}\cdot\text{Azimuthal}\cdot\text{Vertical}\cdot\text{Perturbation}\cdot\left[\begin{array}{l}\text{Exposure value with uncertainty}\\ \text{for each surveillance capsule}\end{array}\right]$$

In this regard, neutronic exposures are needed for all the "limiting" weld or other materials; the "beltline region of the reactor vessel" is defined as encompassing indeed any weld or materials for which the predicted adjustment of reference temperature at the end of its service life exceeds 50°F.[13]

The vertical correction is derived from dosimetry traverses within the surveillance capsule or from 2D(R,Z) transport theory when the limiting material is significantly outside the vertical range of the dosimeters. It is noted that uncertainties of $\sim$10% or less may arise within the vertical range of the active fuel. This problem becomes more difficult for support structures and is particularly important in the case of water shield tanks (Maine Yankee, Connecticut Yankee, Surry, BR3) for which the NDT temperature may be elevated by irradiation to equal or even exceed the service temperature.[57,93] This will be addressed as part of the NESDIP Program.[29,61]

Benchmarking the neutron field perturbation by the surveillance capsule and RM sensor counting laboratory certification tests is an important part of the ORR-SDMF program. As previously shown in Figures 3.16 and 3.17, significant results have already been obtained for Westinghouse and Combustion Engineering type capsules; and the ORR-SDMF irradiation is complete for Babcock and Wilcox type capsules (Figure 3.18). Also, results of recent service laboratory RM sensor counting certification tests for four reactor vendors and two other service laboratories in the U.S. and four laboratories in Europe were previously presented in Tables 3.6, 3.7, and 3.8 where HEDL and CEN/SCK served as the reference counting laboratories for these tests, respectively. It was further stated that RM sensor counting results in the 5 to 10% (1σ) range must be obtained routinely to achieve derived exposure parameter values (fluence $E > 1.0$ MeV, dpa, etc.) in the 10 to 20% (1σ) range desired for fracture analysis studies.

The radial in-vessel projection, exclusive of surveillance capsule perturbation effects, has been addressed by the PCA blind test[23] and is reasonably well understood. Three main areas of discrepancies or inconsistencies remain:

- Integral C/E ratios at deep penetration and high neutron energy indicate that calculations underpredict the flux; this is traced to iron cross-section inadequacies in current nuclear data files.[23,58]

- Differences between fission chamber and SSTR[23] measurement results have been observed; further benchmark-field referencing work is expected to largely resolve this problem, see Section 2.4.3.

- Neutron spectrometry versus integral measurement and calculation studies are in progress: Comparison of current transport theory with the envelope (Figure 3.34) of all ^{6}Li(n,α) energy-dependent flux spectrum attenuations as a function of steel penetration (PCA 8/7 and 12/13, 1/4 T versus 1/2 T, and 1/2 T versus 3/4 T ratios) displays overall trends compatible with the ones under Figure 3.33, but inconsistencies are claimed at the level of more detailed confrontations.[23]

Figure 3.33 was also prepared to illustrate the transferability of neutronic benchmark observations to power reactor environments. From an applied RPV engineering viewpoint, the primary program goals have been reached; R&D improvement of the current PCA blind test results is not considered a high priority, but should be useful for: (a) the analysis of pressurized thermal shock insofar as more accurate dpa steel traverses would ensue (the critical crack arrest depth after initiation of shallow flaws is relatively sensitive to these traverses, but a host of other uncertainties may be more critical at present); and (b) the interpretation of ex-vessel physics-dosimetry, both in the context of a better understanding of lead factor uncertainties and in assessing support structure embrittlement.[57,93]

The benchmarking of azimuthal neutron flux spectrum gradient predictions for in-vessel locations is addressed in the VENUS zero-power engineering mockup of a PWR core-baffle-barrel-thermal shield configuration, see Figure 3.43.[18,60] These predictions depend on:

- Correct and detailed estimates of core fission source distributions in the last core fuel rows relative to the plant power output.

- Correct modeling of core boundary heterogeneity effects.

The first aspect is a particularly important focus for investigation because usual core management considerations do not call for an accuracy as great as needed for in-vessel RPV surveillance projections. Current lead factor uncertainties are, therefore, likely to be dominated by core fission source uncertainties and are likely to be the most significant in plants displaying large azimuthal effects (Westinghouse, Combustion Engineering); these effects are not (or are less) sensitive to fuel burnup,[59] which enhances the value of results from a zero-power benchmark. On another hand, in-vessel azimuthal gradients are attenuated by scattering within the vessel and distorted by the cavity. This may be related to vessel exposure [fluence ($E > 1.0$ MeV) and dpa] when sufficient data and techniques are available from benchmark and in-reactor tests, see Section 3.4.3.1.3 and References 18, 23-29, 31-38, 40-43, 45-51, 53-56, 58-61, 66, 72, 73, 76, 78, 80. The VENUS and NESDIP programs are expected to provide verification for in-vessel azimuthal gradient calculations and a better understanding and verification of in- and ex-vessel neutron and gamma field predictive methods. Thus, the VENUS and NESDIP programs will contribute to the development, testing, and verification of a fracture analysis predictive methodology for RPV application and ex-vessel dosimetry, which otherwise could never become quantitative and comprehensive. Two other essential aspects of the VENUS effort, as already discussed in Ref. 18, are the investigation of pressurized thermal shock mitigation by core management techniques and the investigation of PWR gamma heating.

Further discussion of the VENUS and NESDIP programs is provided in References 60 and 61, respectively. It is useful to mention that the experimental and analytical program is interlaboratory and open to more participants than the ones already engaged in the U.S., Belgium and the

United Kingdom. In this regard, the active participation of reactor vendors, architect/engineers, and utilities is deemed essential.

3.5. CONCLUSIONS AND REGULATORY DEMANDS

From the above it is concluded that fuel management schemes provide practical and perhaps relatively inexpensive ways of reducing or eliminating the risk of PV fracture associated with pressurized thermal shock. Assessment and control of the conditions of LWR pressure vessels and support structures are related problems. The regulatory demand[7] is for assurance (verification)

1) that errors in neutron exposure values (fluence E > 1.0 MeV) of a factor of two are a thing of the past; i.e., that there are no more technical surprises, for instance, due to a lack of knowledge of the effects of old and new fuel management schemes,

2) that an improved neutron exposure parameter (such as dpa) be used to account for neutron spectral effects,

3) that gamma heating be better understood to account for steel metallurgy time-temperature effects, and

4) that all of the physics-dosimetry-metallurgy information correlates properly with the embrittlement of the reactor vessel and support structure materials.

To meet the above challenge, a new series of ASTM standards is being developed, tested, verified, and applied for LWR pressure vessel and support structure surveillance.[17,74,94] It is expected that all of these standards will be in place by late 1985, with appropriate revisions thereafter. Routine and careful application of these recommended ASTM physics-dosimetry-metallurgy methods, procedures and data will allow verification at the required accuracy level (10 to 30%, 1σ) of the effects of old and new fuel management schemes on the estimated current and end-of-life condition of pressure vessel and support structure steels.

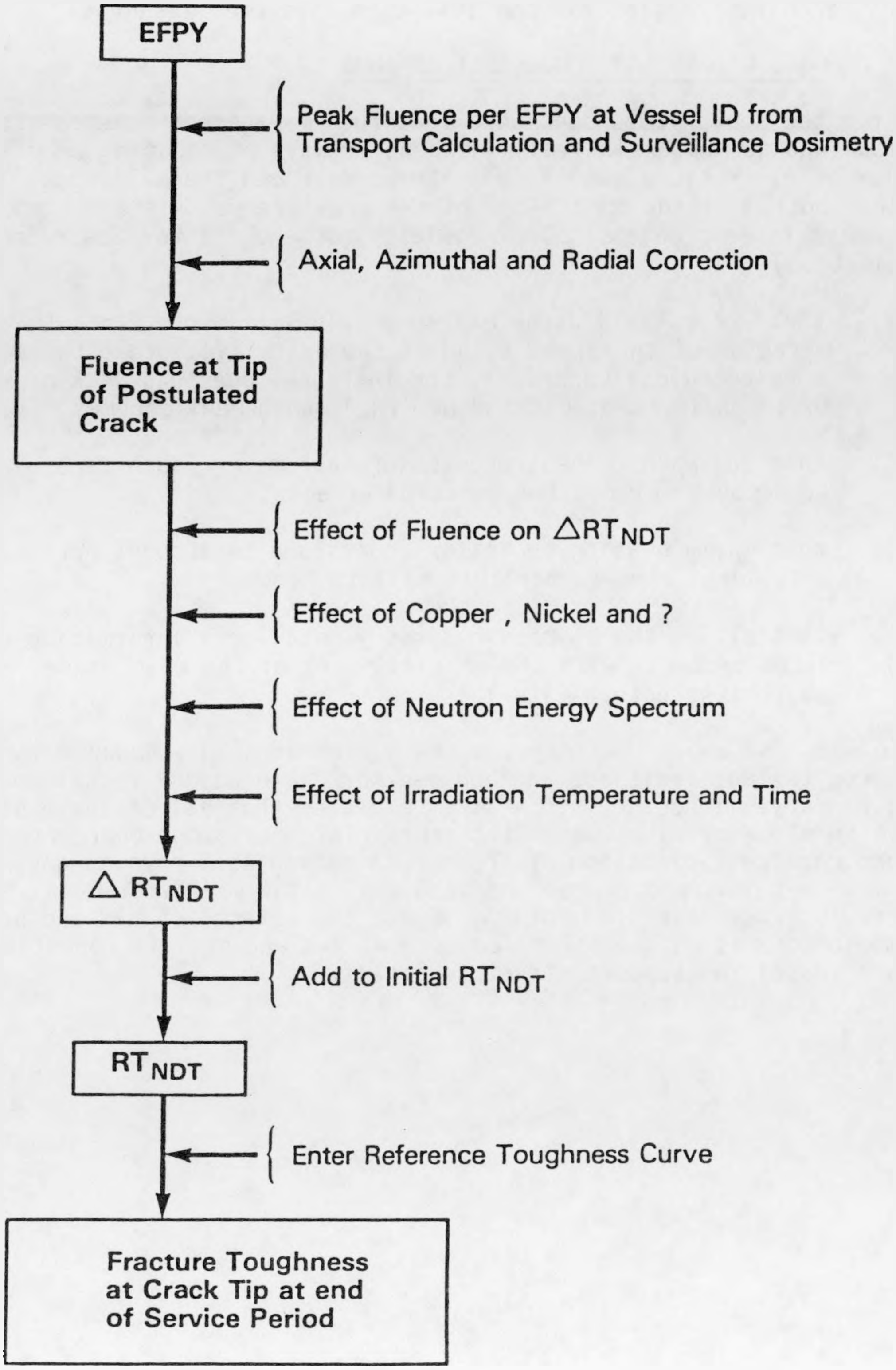


FIGURE 3.1. Flow Sheet for the Application of Neutron Physics-Dosimetry-Metallurgy to Fracture Analysis (taken from Reference 7).

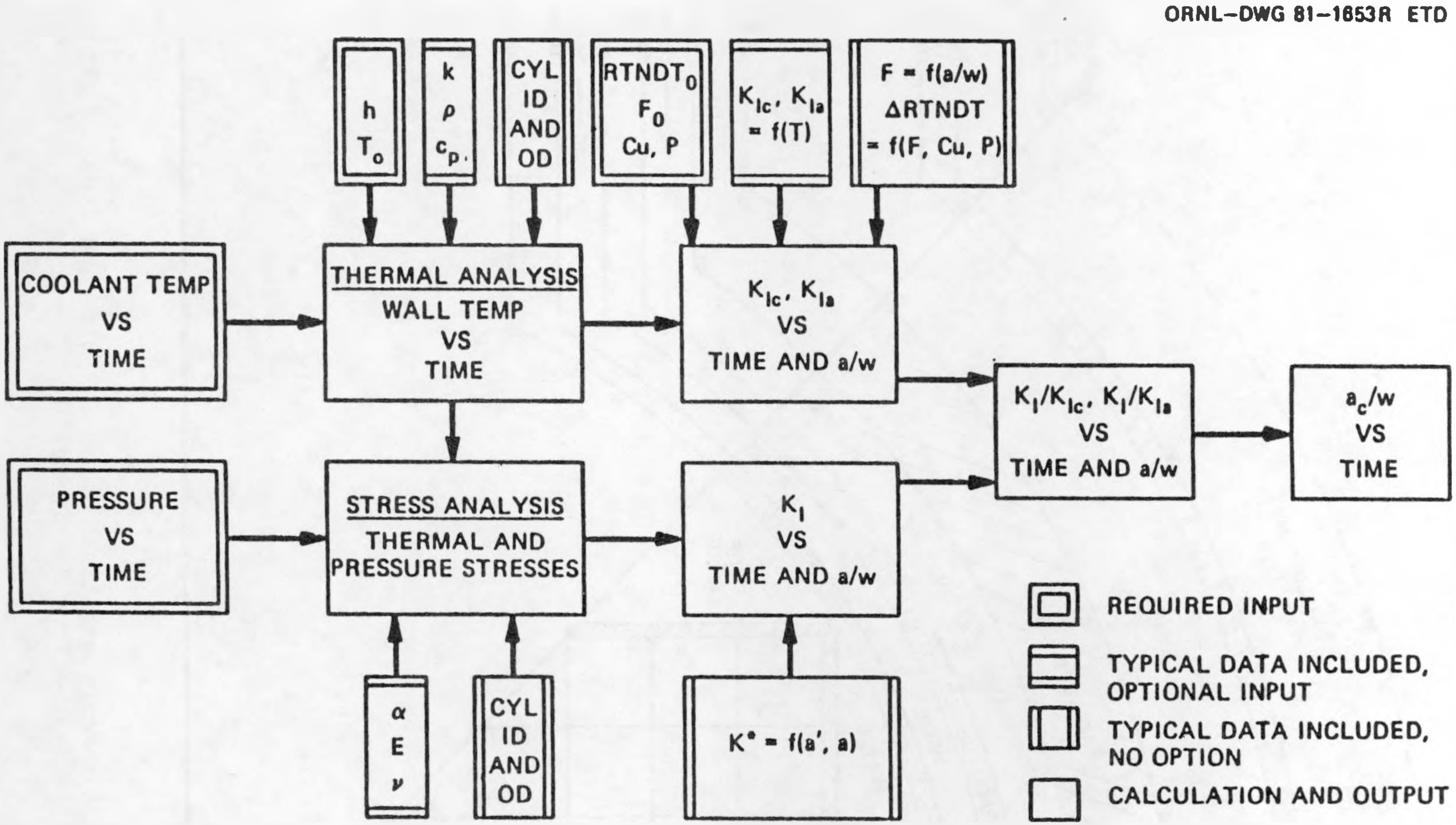


FIGURE 3.2. Block-Diagram Description of OCA-I, Indicating Basic Input, Calculations and Output (taken from Reference 20).

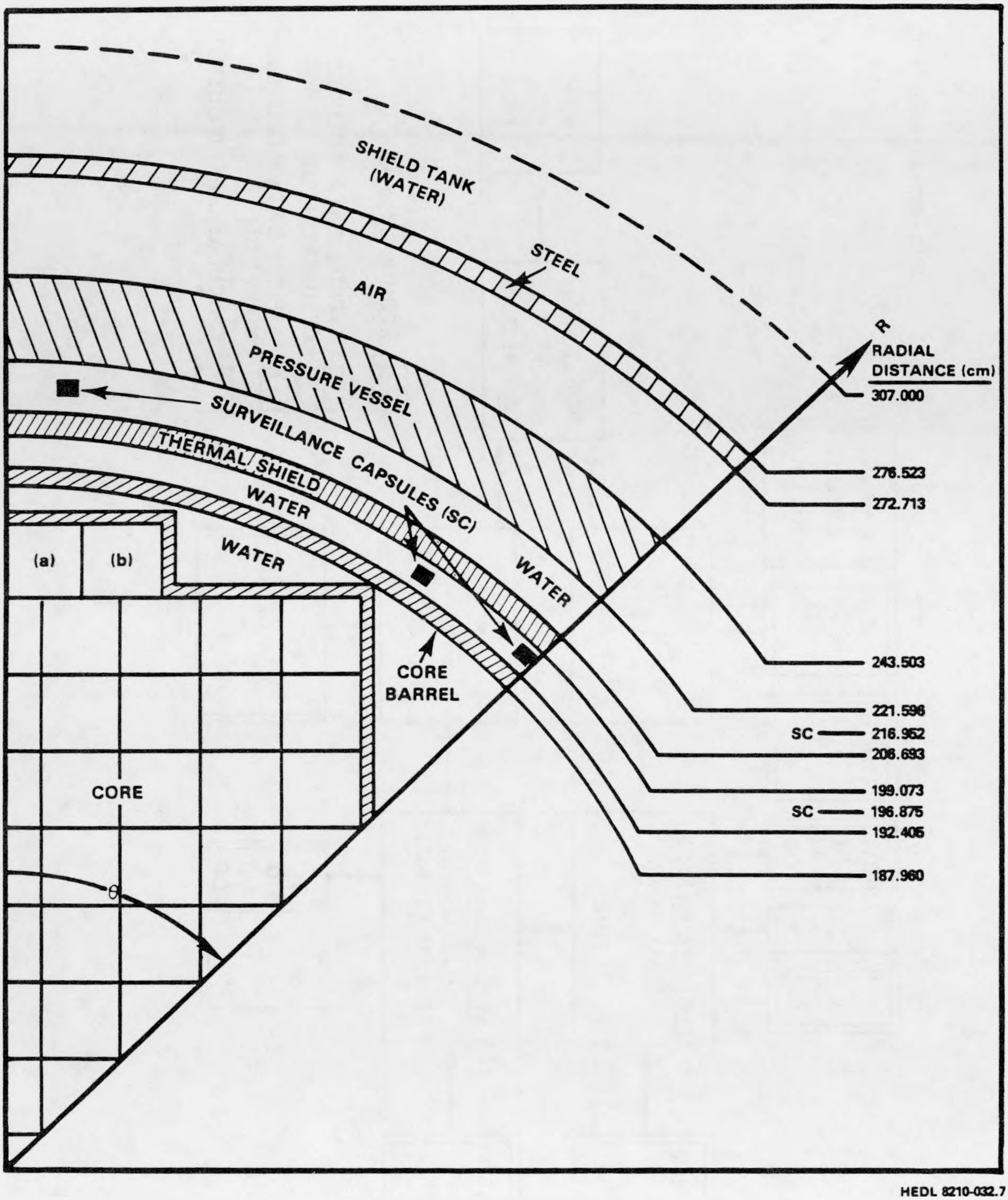


FIGURE 3.3. Schematic Representation Type A PWR with Two Types of Surveillance Capsules (taken from Reference 10).

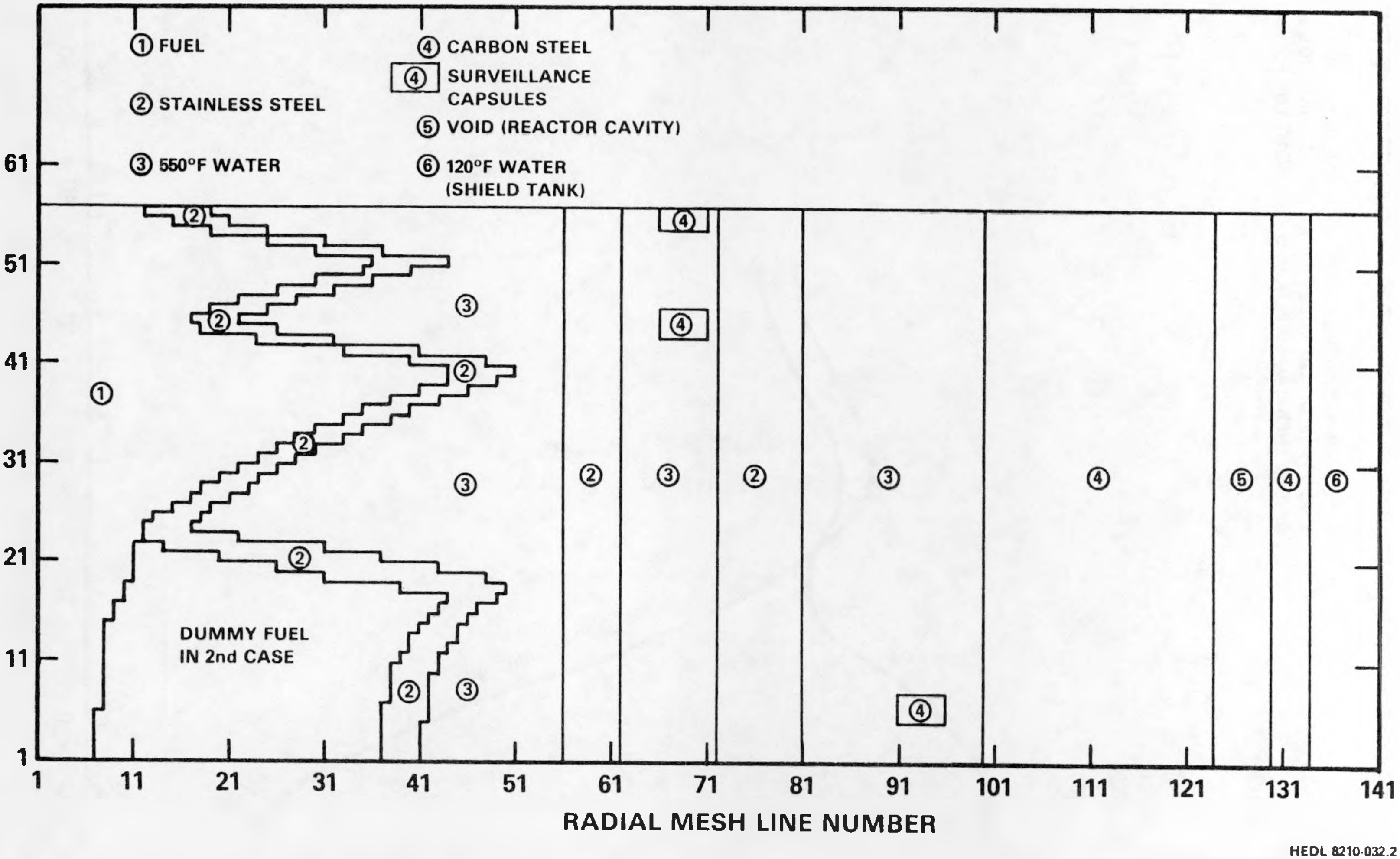


FIGURE 3.4. Mesh Line Description for R,θ Analysis of the Type A Reactor with Two Types of Surveillance Capsules (taken from Reference 10).

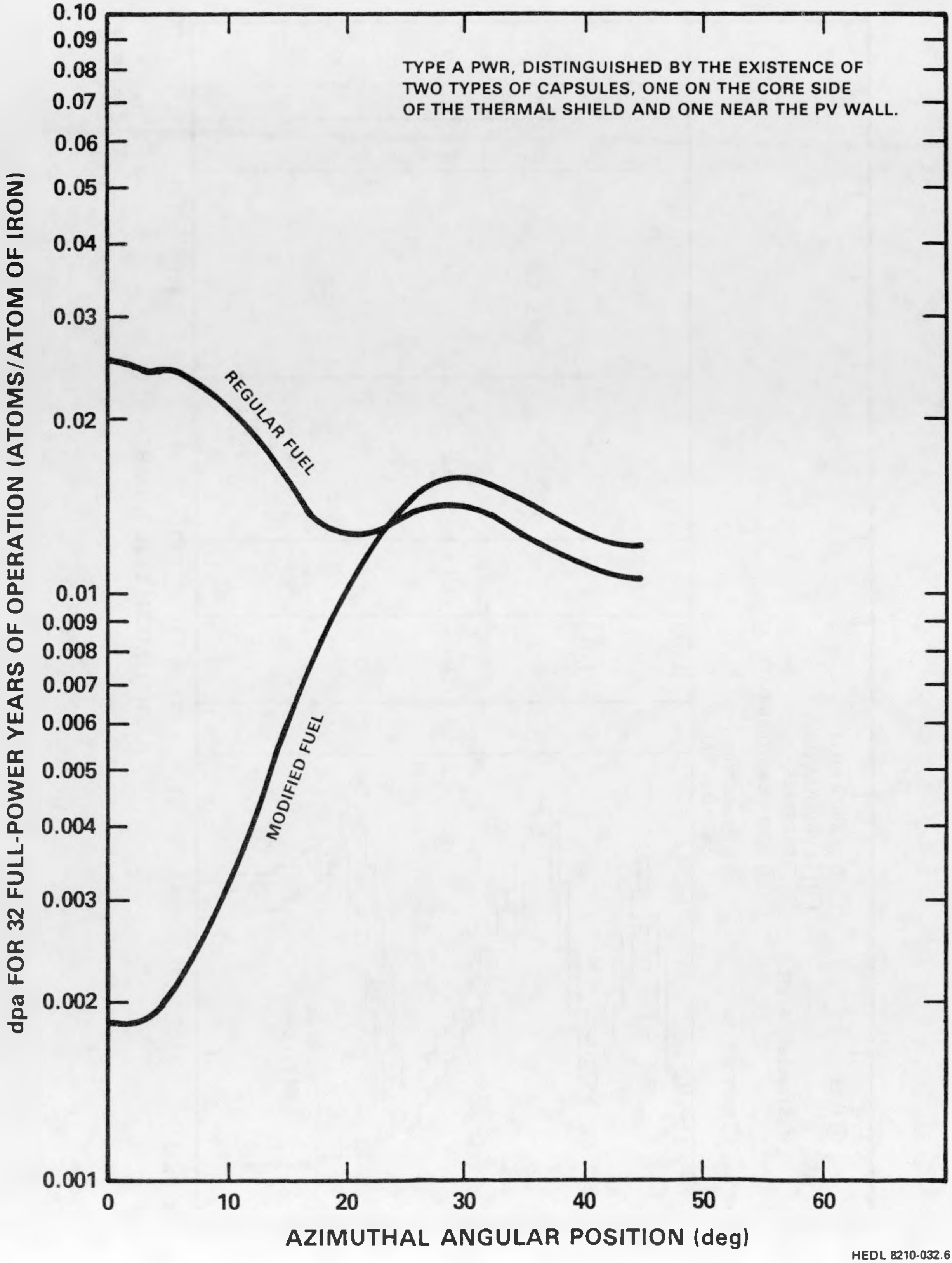


FIGURE 3.5. Dpa for 32 Years Full-Power Exposure on the Front Face of the Pressure Vessel, Plotted as a Function of Angular Position (taken from Reference 10).

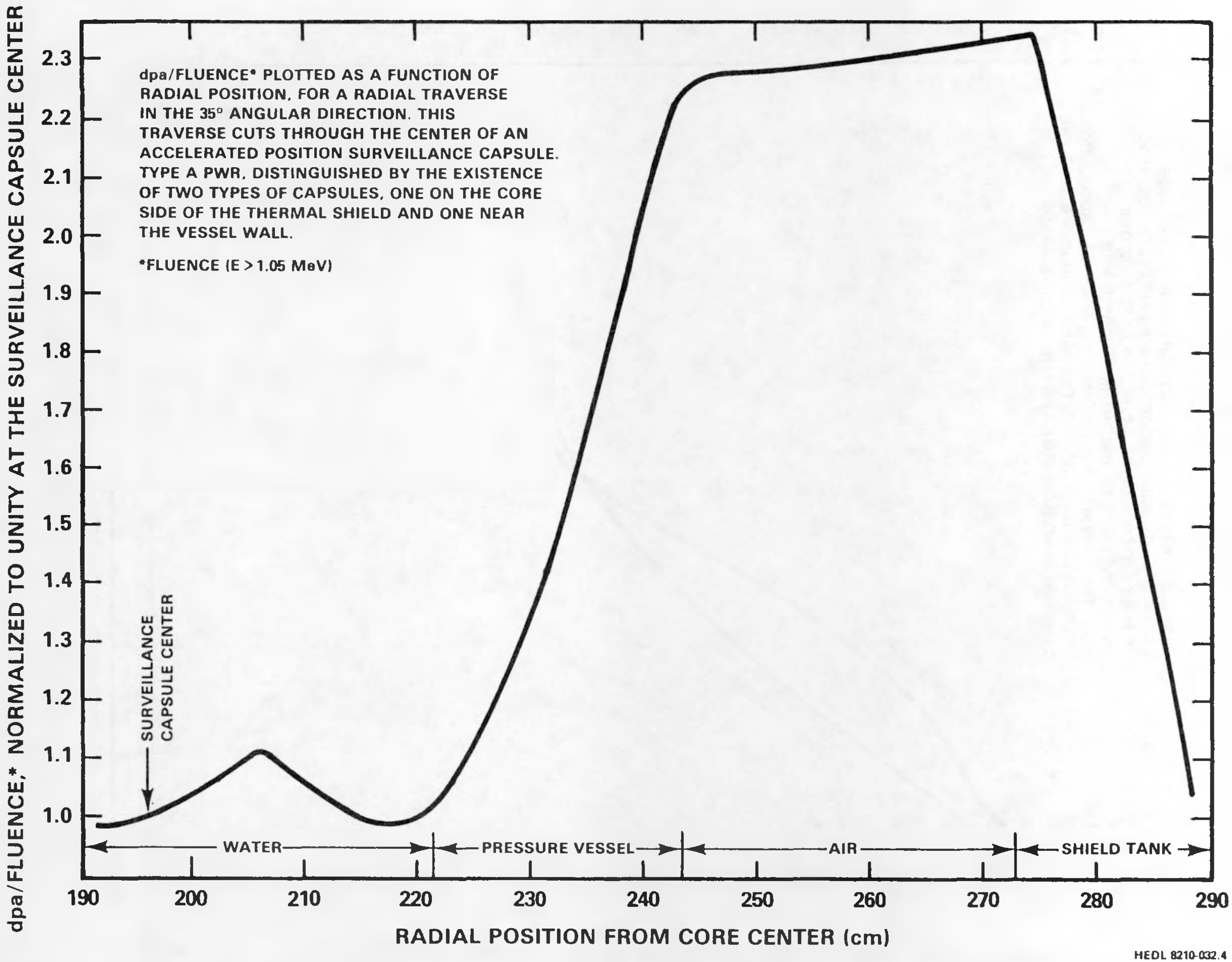


FIGURE 3.6. Dpa/Fluence Radial Traverse for Type A PWR (taken from Reference 10).

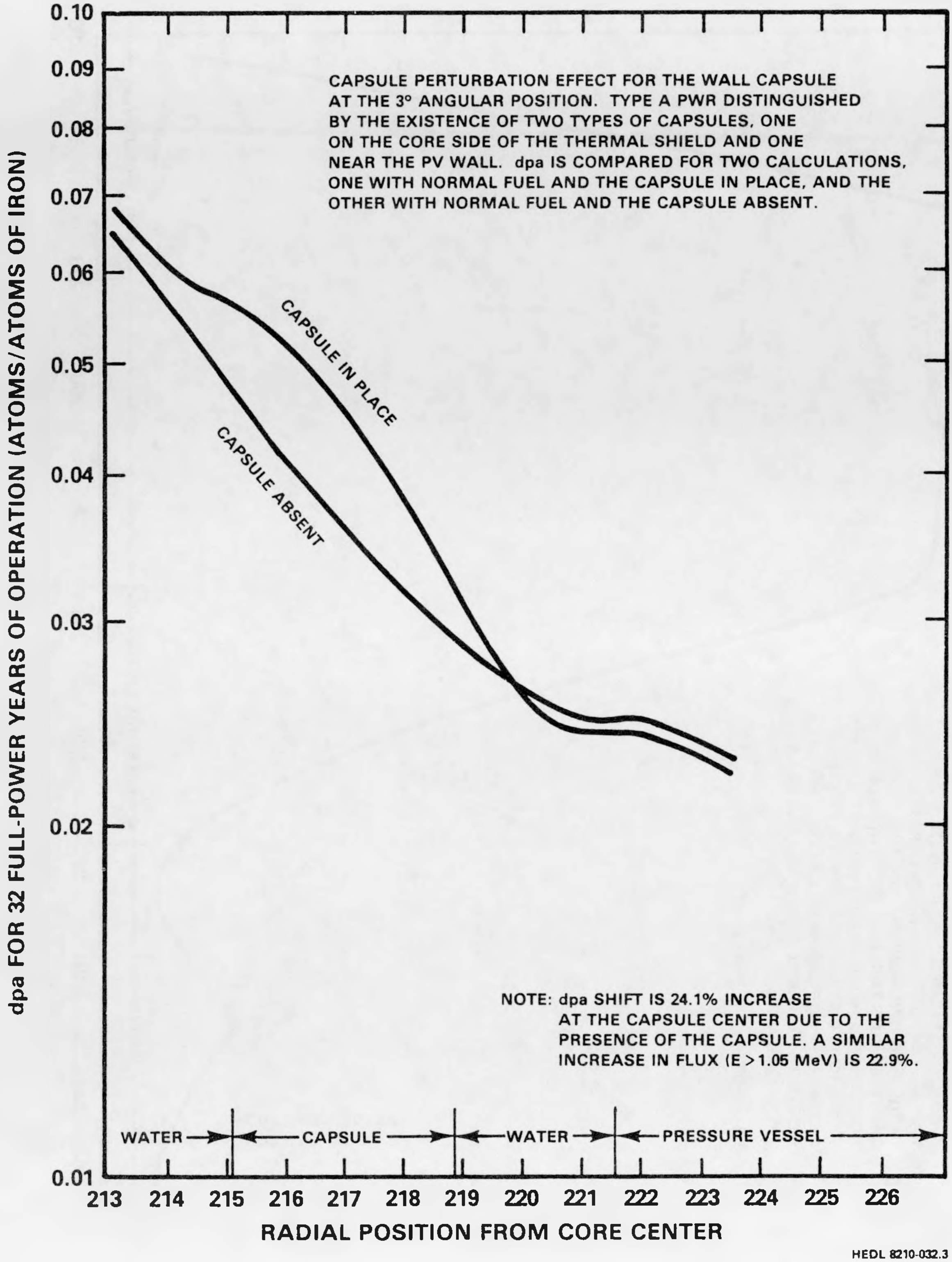


FIGURE 3.7. Wall Capsule Perturbation Effect for Type A PWR (taken from Reference 10).

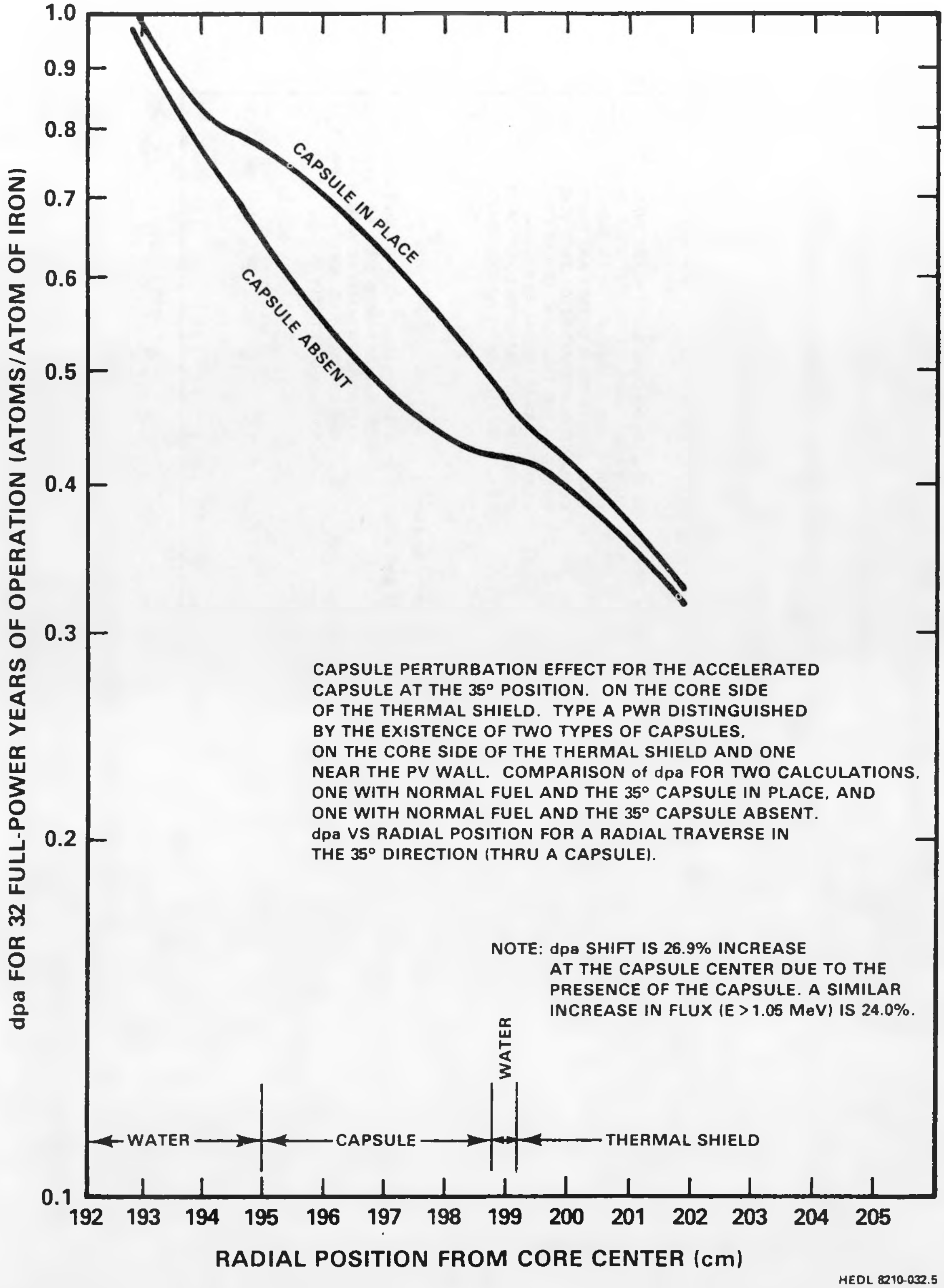


FIGURE 3.8. Accelerated Capsule Perturbation Effect for Type A PWR (taken from Reference 10).

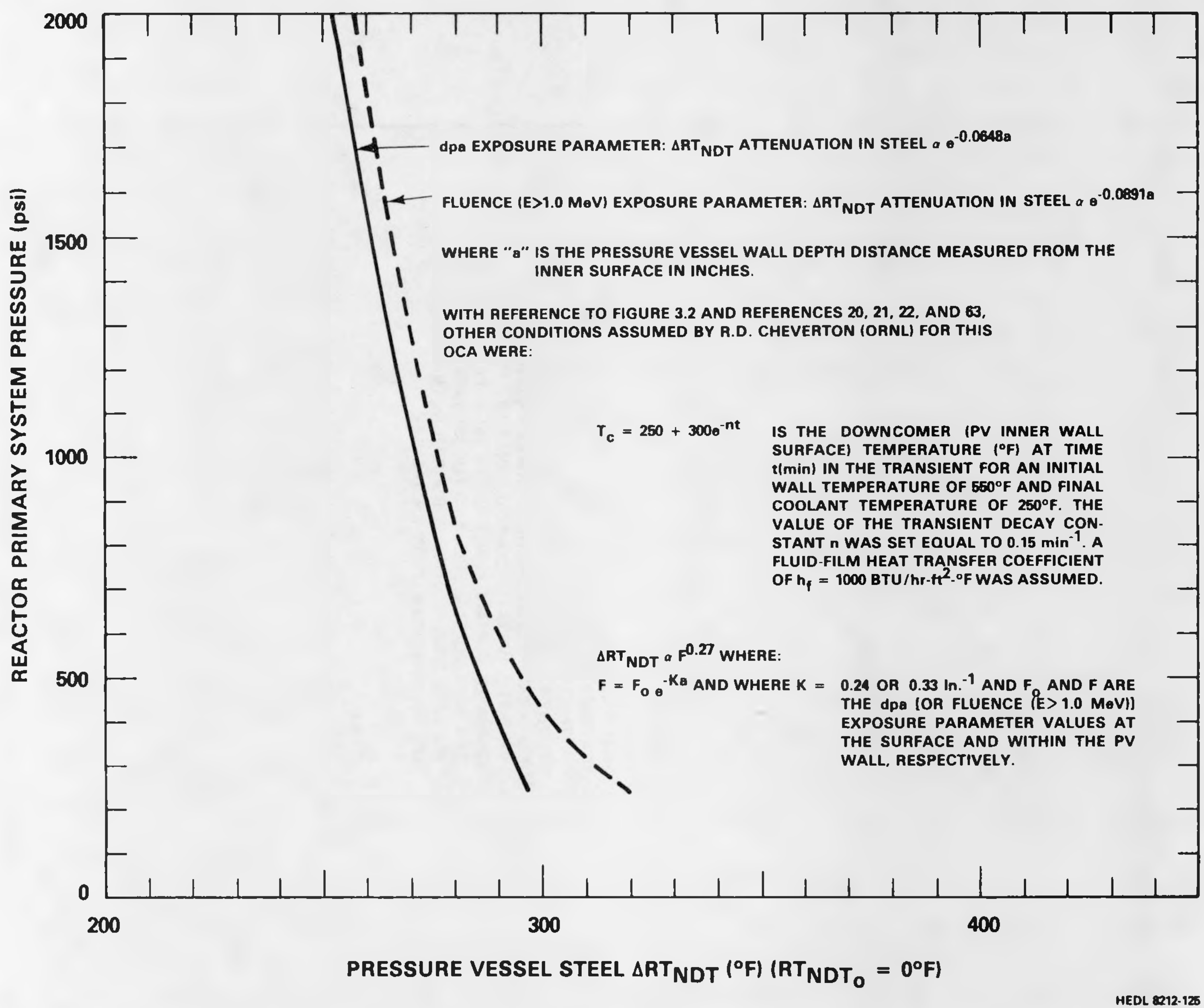
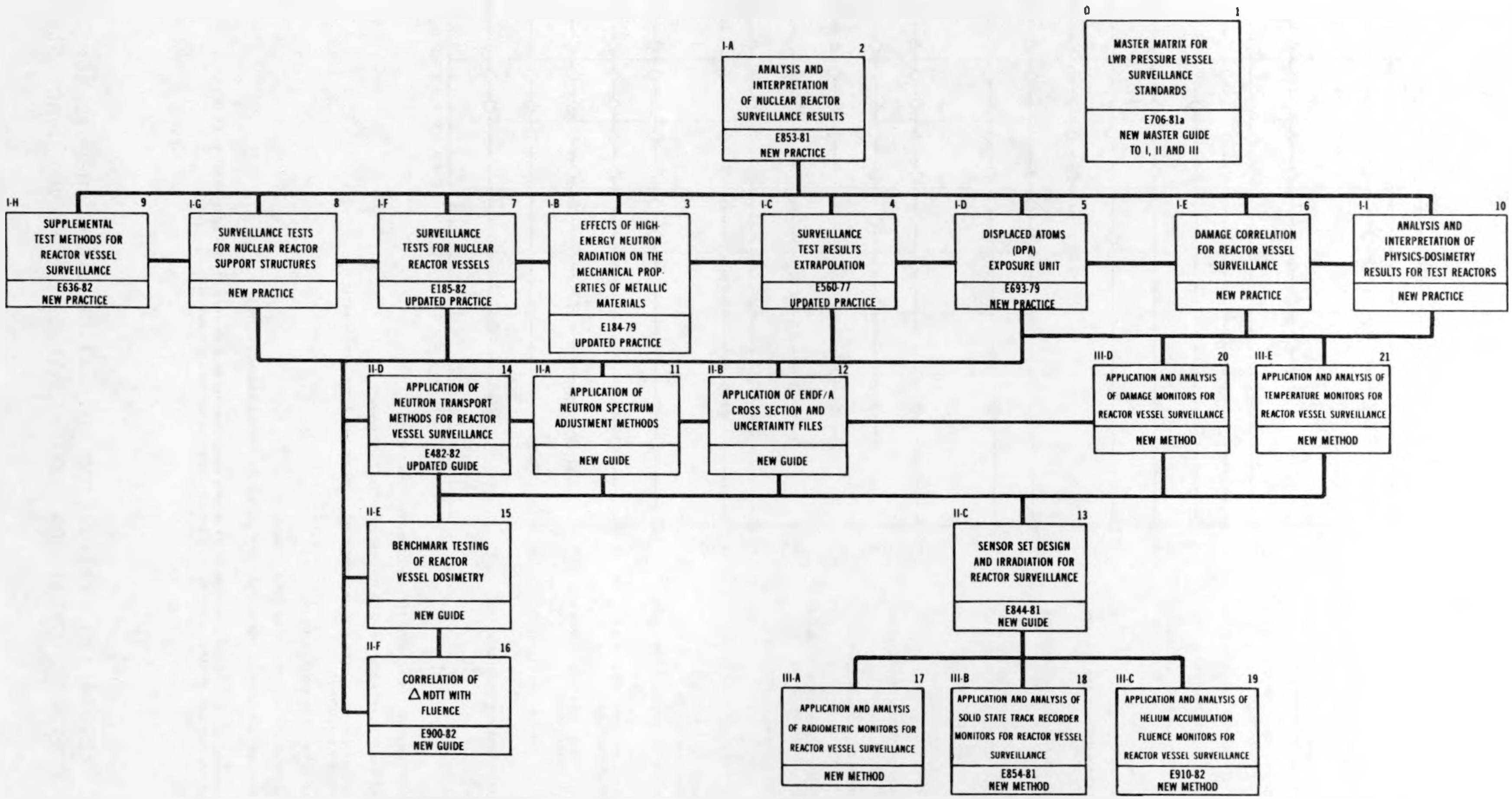


FIGURE 3.9. Maximum Pressure to Permit Crack Arrest for a Postulated Overcooling Accident Using the OCA-I Code with Two Different Exposure Parameters for the Attenuation of ΔRT_{NDT} in Steel.



ASTM STANDARDS FOR SURVEILLANCE OF LWR NUCLEAR REACTOR PRESSURE VESSELS AND SUPPORT STRUCTURES.

FIGURE 3.10. ASTM Standards for Surveillance of LWR Nuclear Reactor Pressure Vessels and Support Structures.

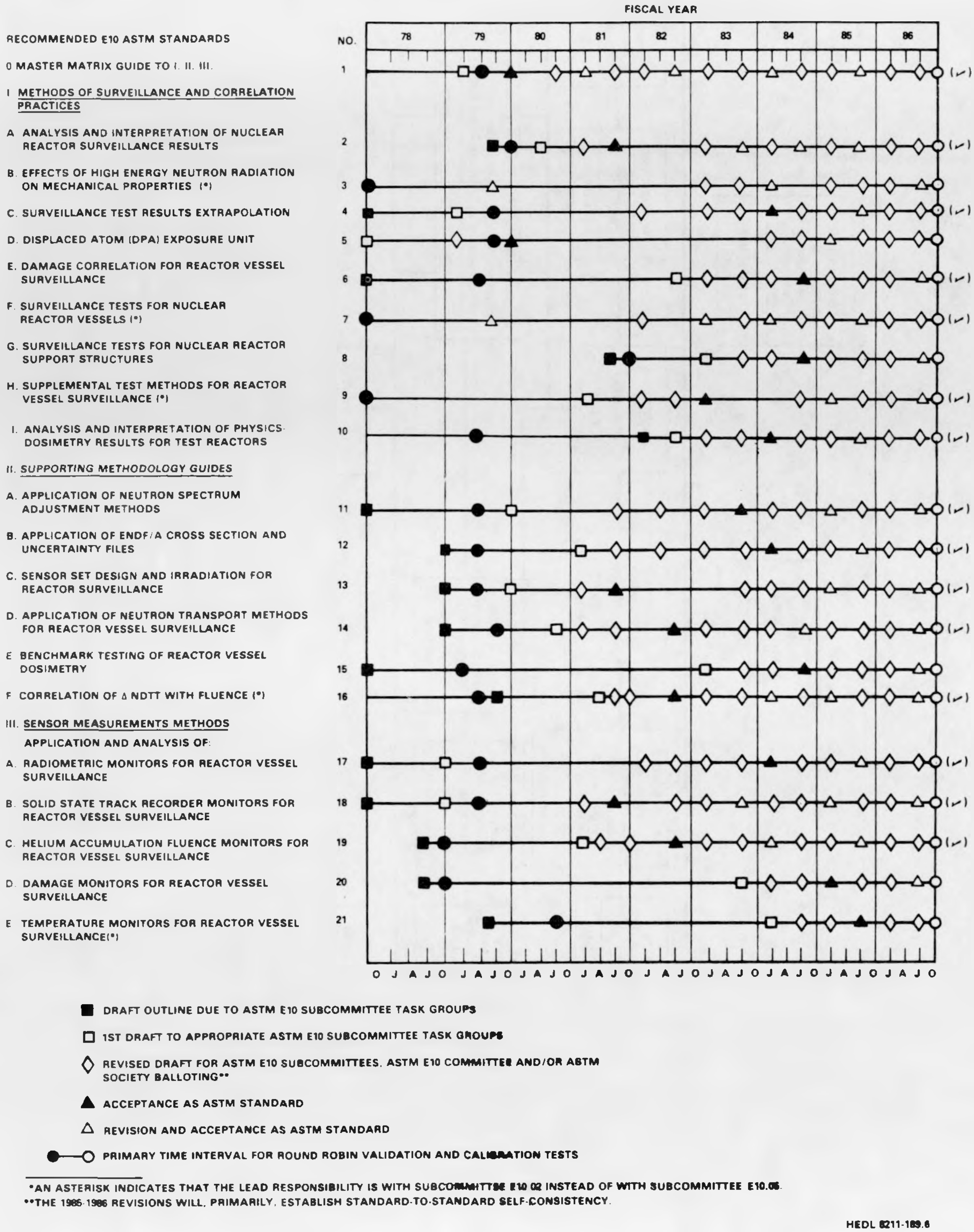


FIGURE 3.11. Preparation, Validation and Calibration Schedule for LWR Pressure Vessel and Support Structure Surveillance Standards.

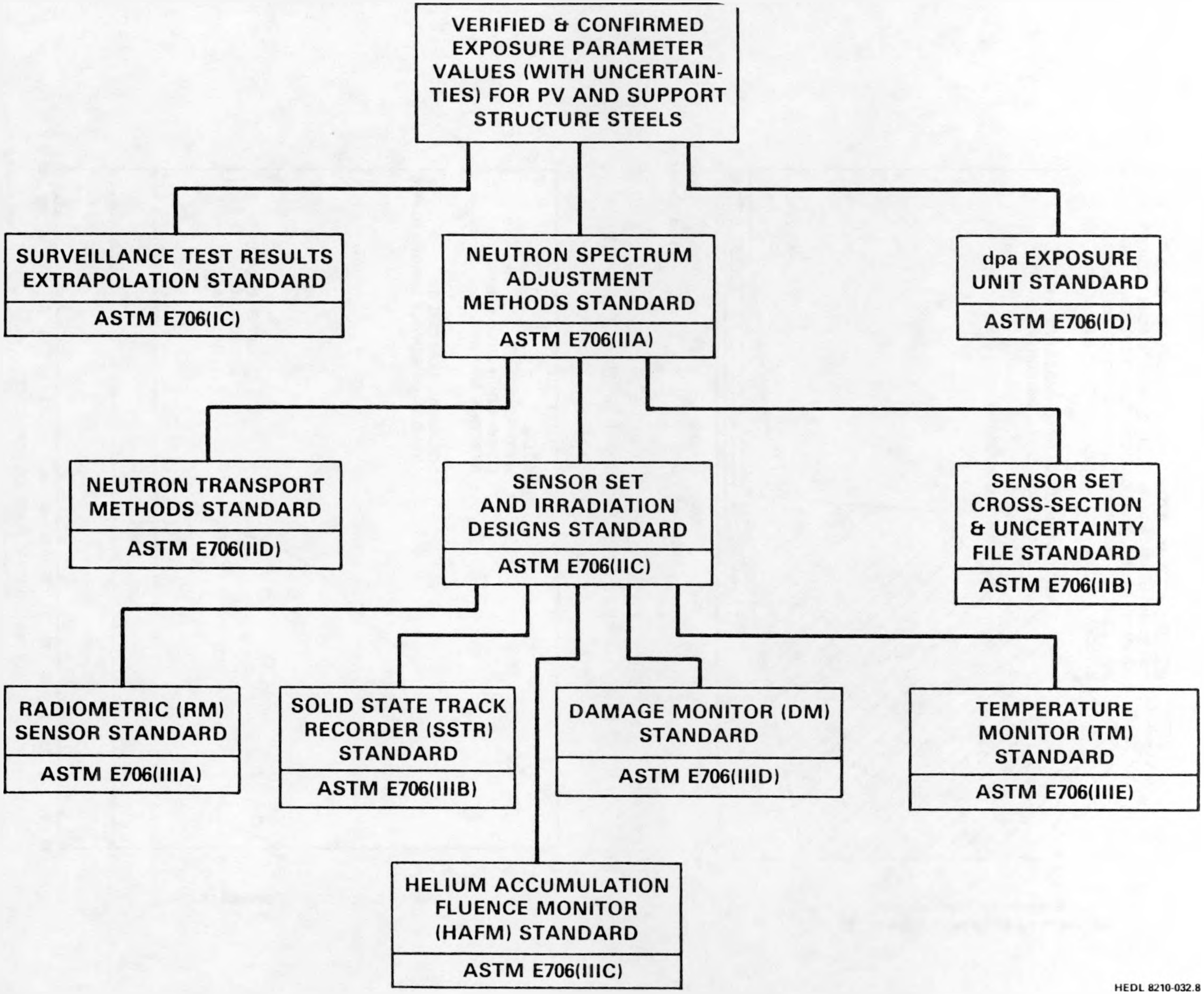


FIGURE 3.12. Interrelationship of ASTM Physics-Dosimetry Standards to Determination of Exposure Values.

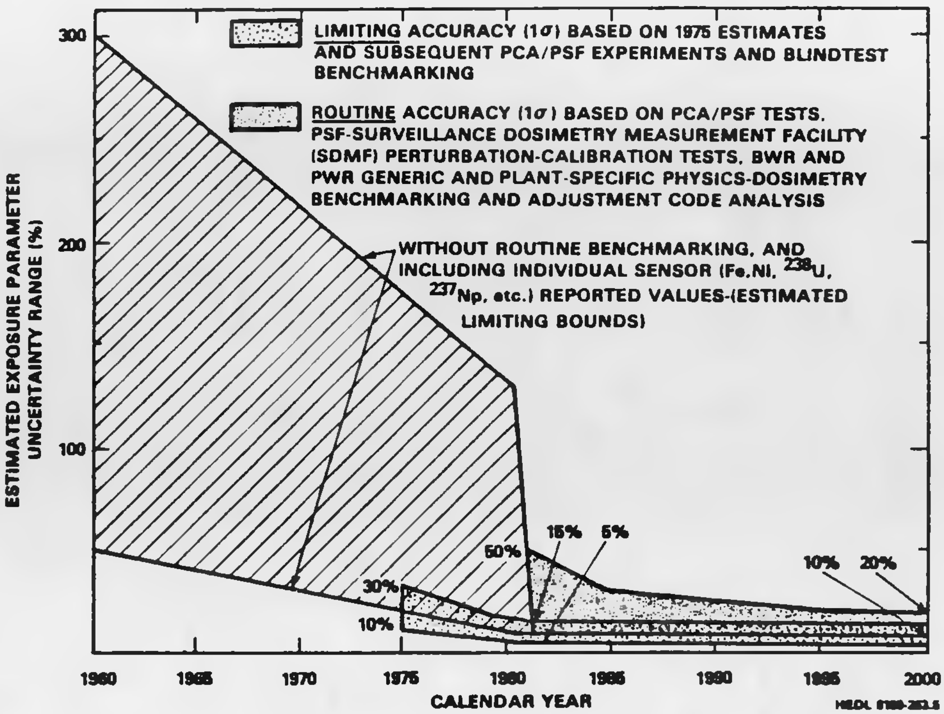


FIGURE 3.13. Estimated Exposure Parameter Uncertainties Obtained from FSAR and Surveillance Capsule Reports (taken from Reference 24).

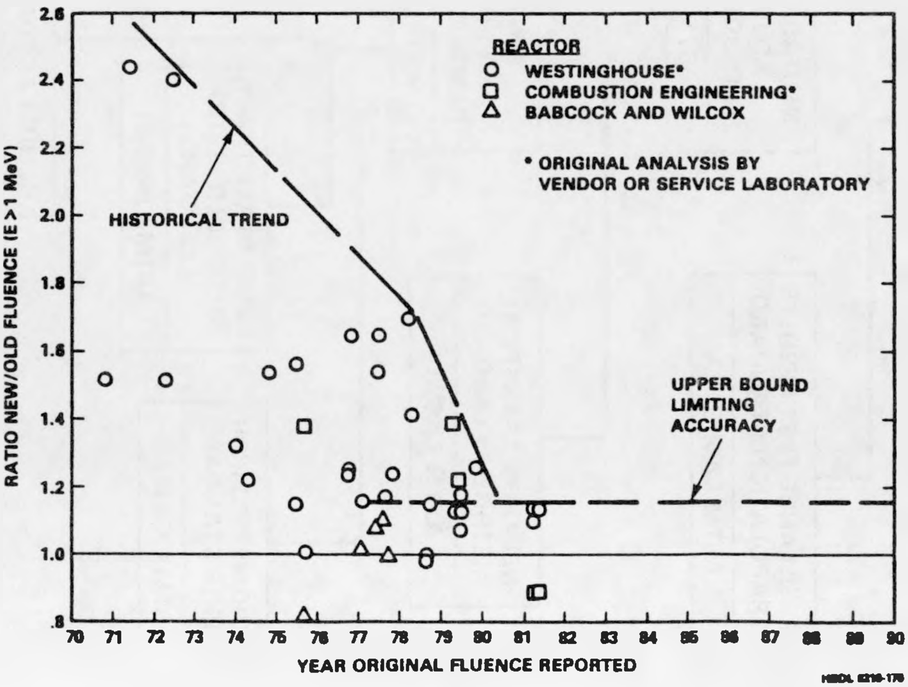


FIGURE 3.14. Ratio of New Fluence/Old Fluence as a Function of Date That Old Fluence was Reported (revision of Reference 40 data).

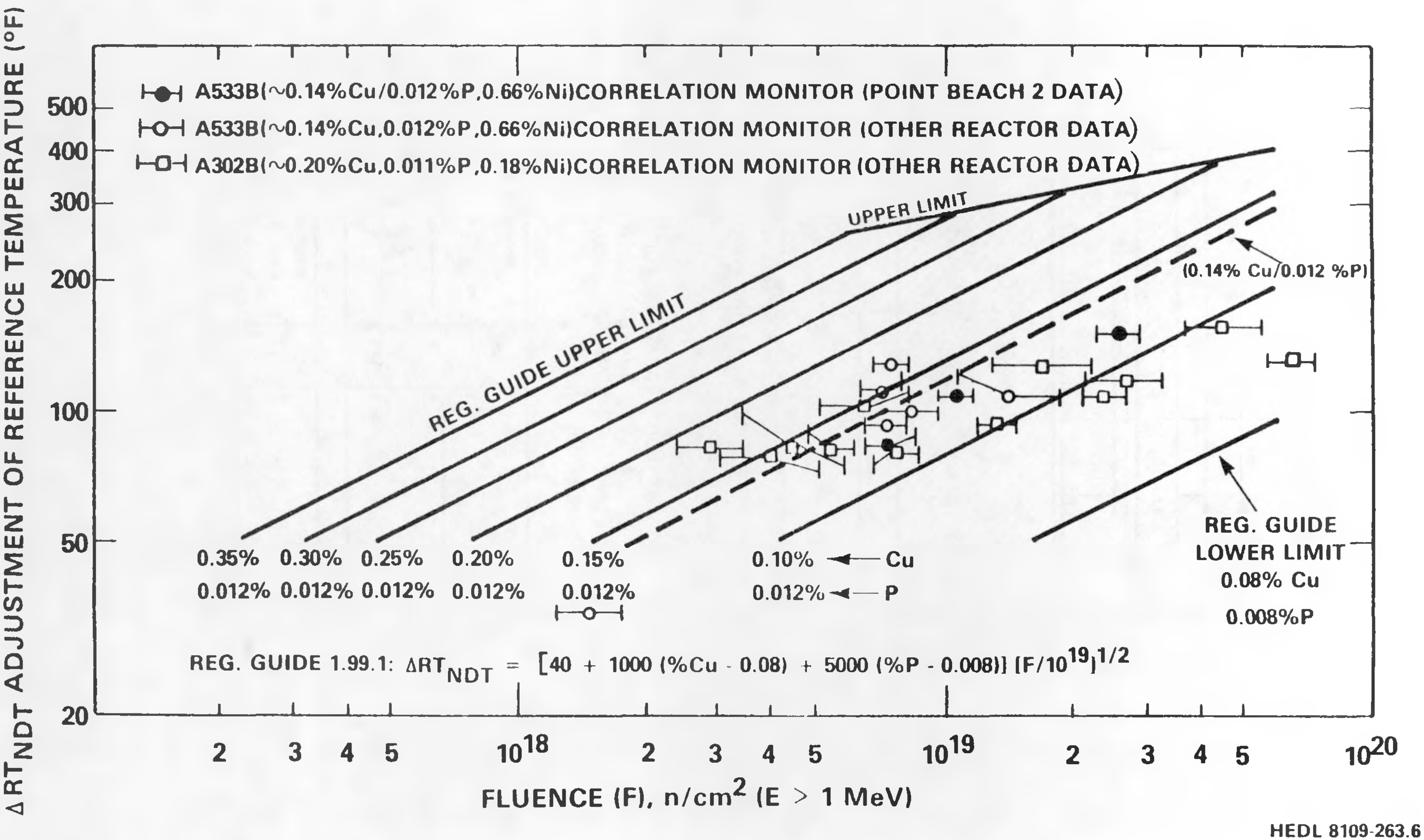


FIGURE 3.15. ΔRT_{NDT} Trend Curve Results for Correlation Monitor Material with New Exposure Values and Uncertainties (taken from Reference 24).

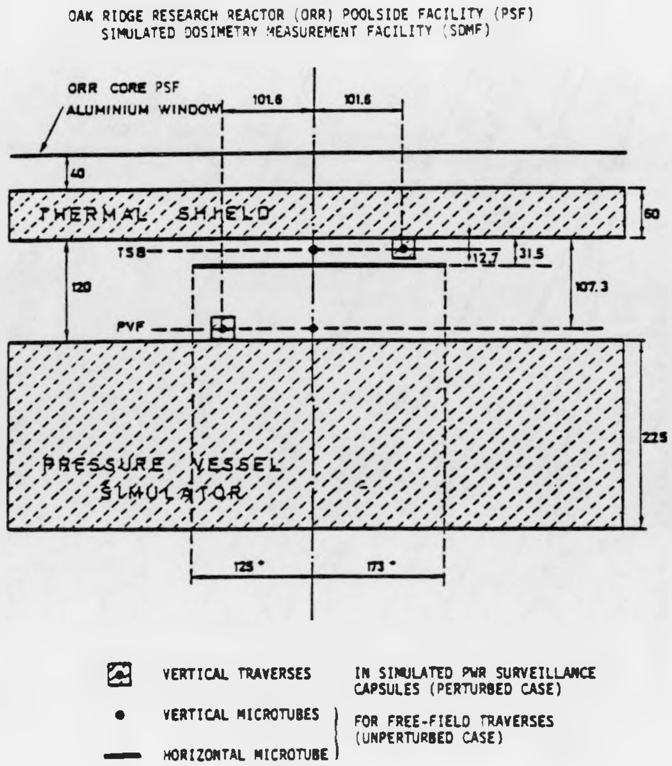


FIGURE 3.16. As-Built Experimental Configuration for (1) Westinghouse and Combustion Engineering Type Surveillance Capsule Perturbation Test and (2) the First ORR-SDMF RM Sensor Certification Test (taken from References 24 and 41).

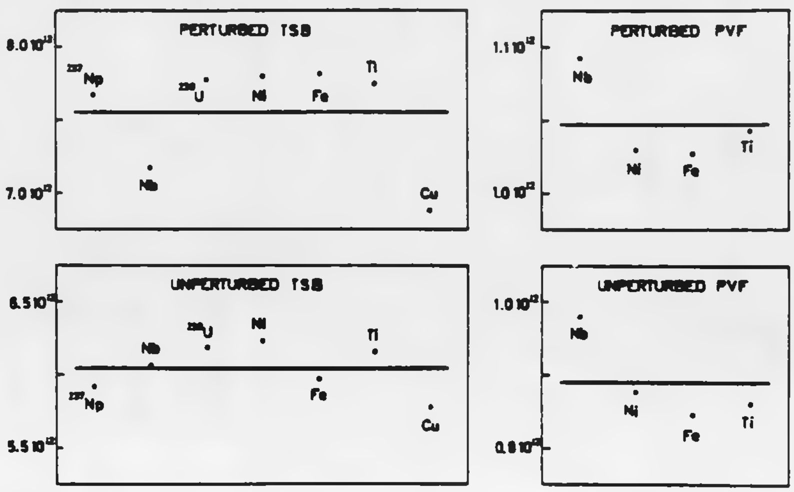


FIGURE 3.17. ϕ (E >1 MeV) in n $s^{-1}cm^{-2}$ at the Thermal Shield Back (TSB) and Pressure Vessel Front (PVF) Positions for the Westinghouse and Combustion Engineering Type Surveillance Capsule Perturbation Test (taken from Reference 41).

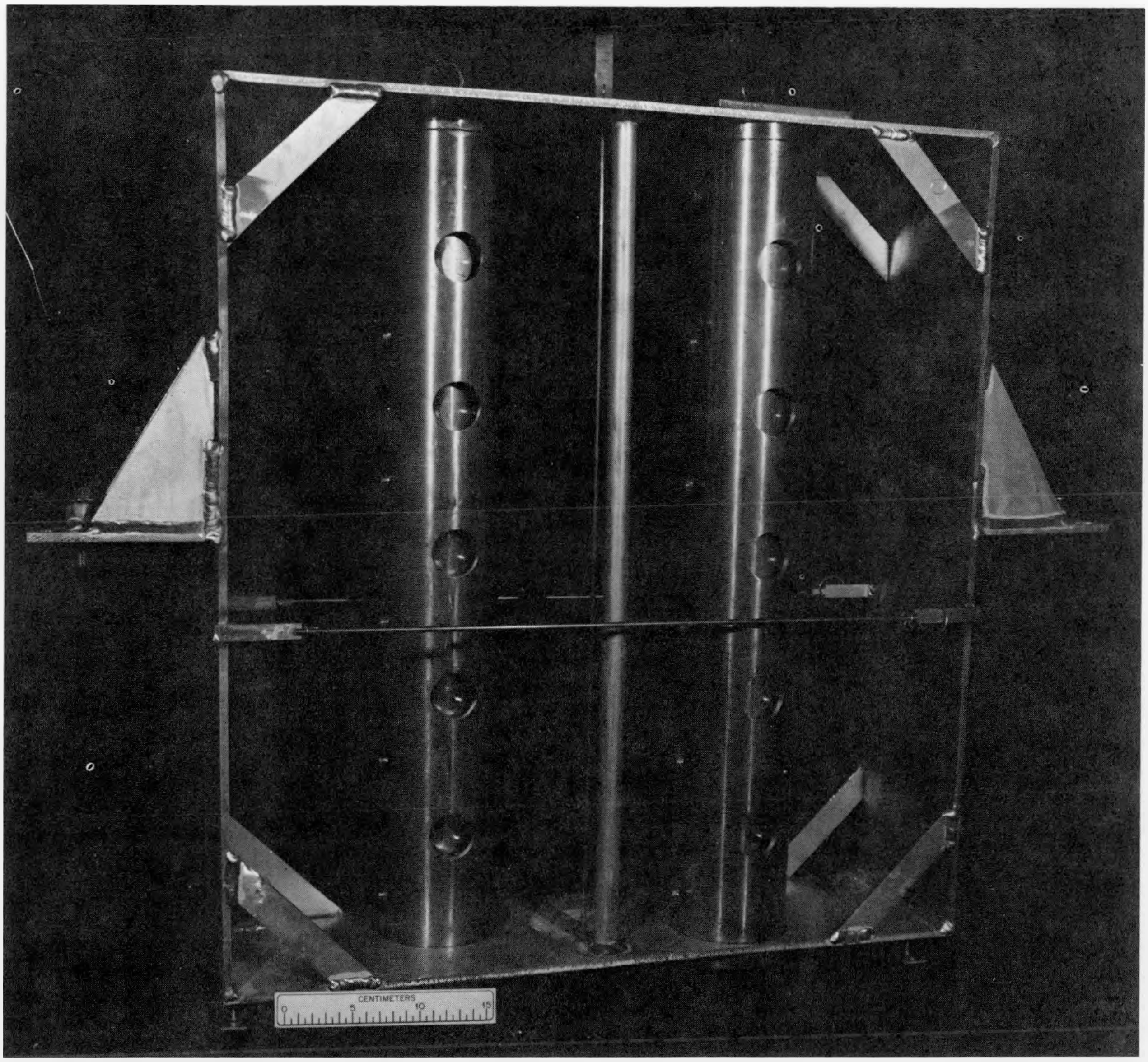


FIGURE 3.18. As-Built Experimental Configuration for the Babcock & Wilcox Type Surveillance Capsule Perturbation Test and Third RM Sensor Certification Test; Fluence (E > 1 MeV) of $\sim 1.0 \times 10^{18}$ n/cm^2.

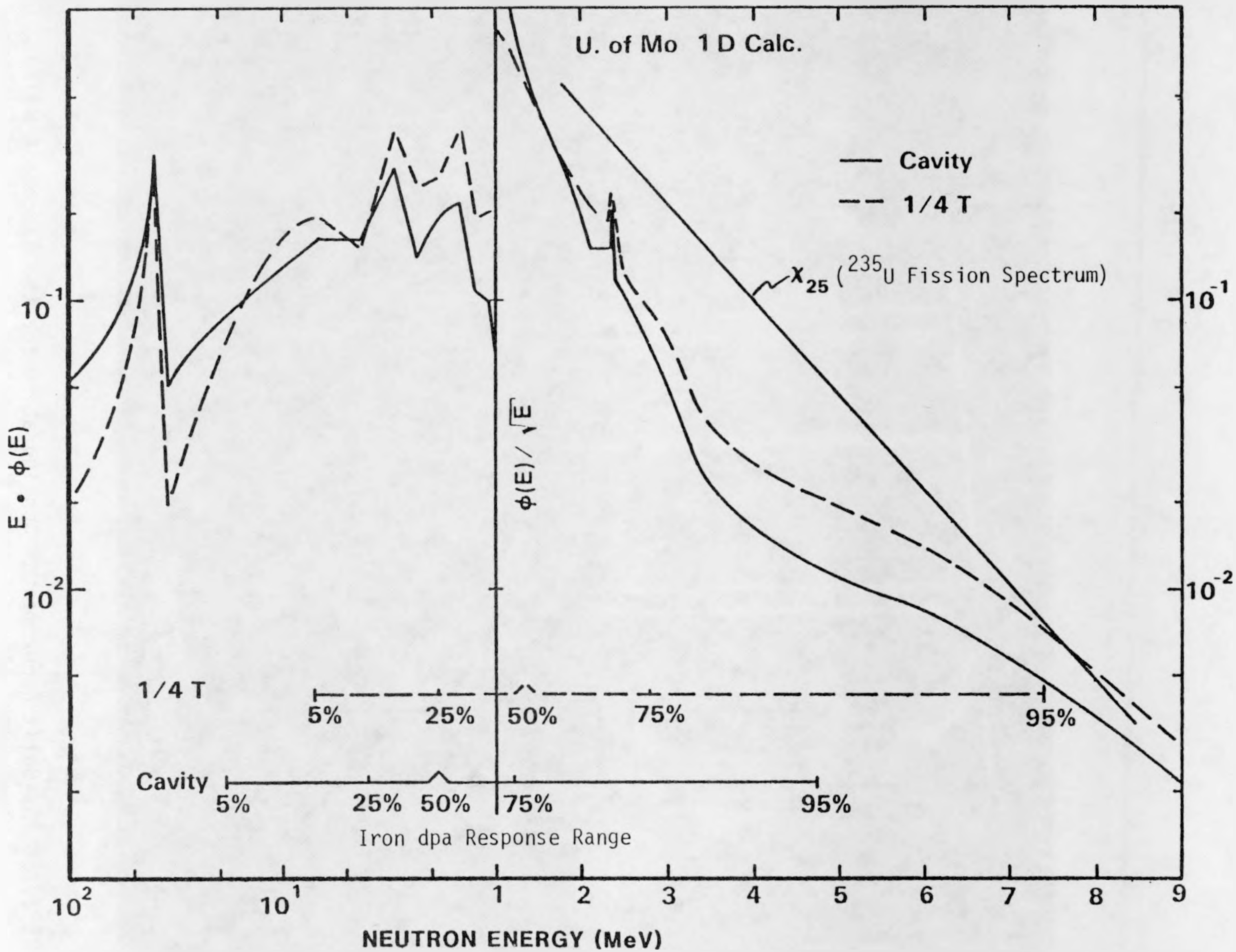


FIGURE 3.19. ANO-1 Cavity and Surveillance Position Neutron Spectra (taken from Reference 32).

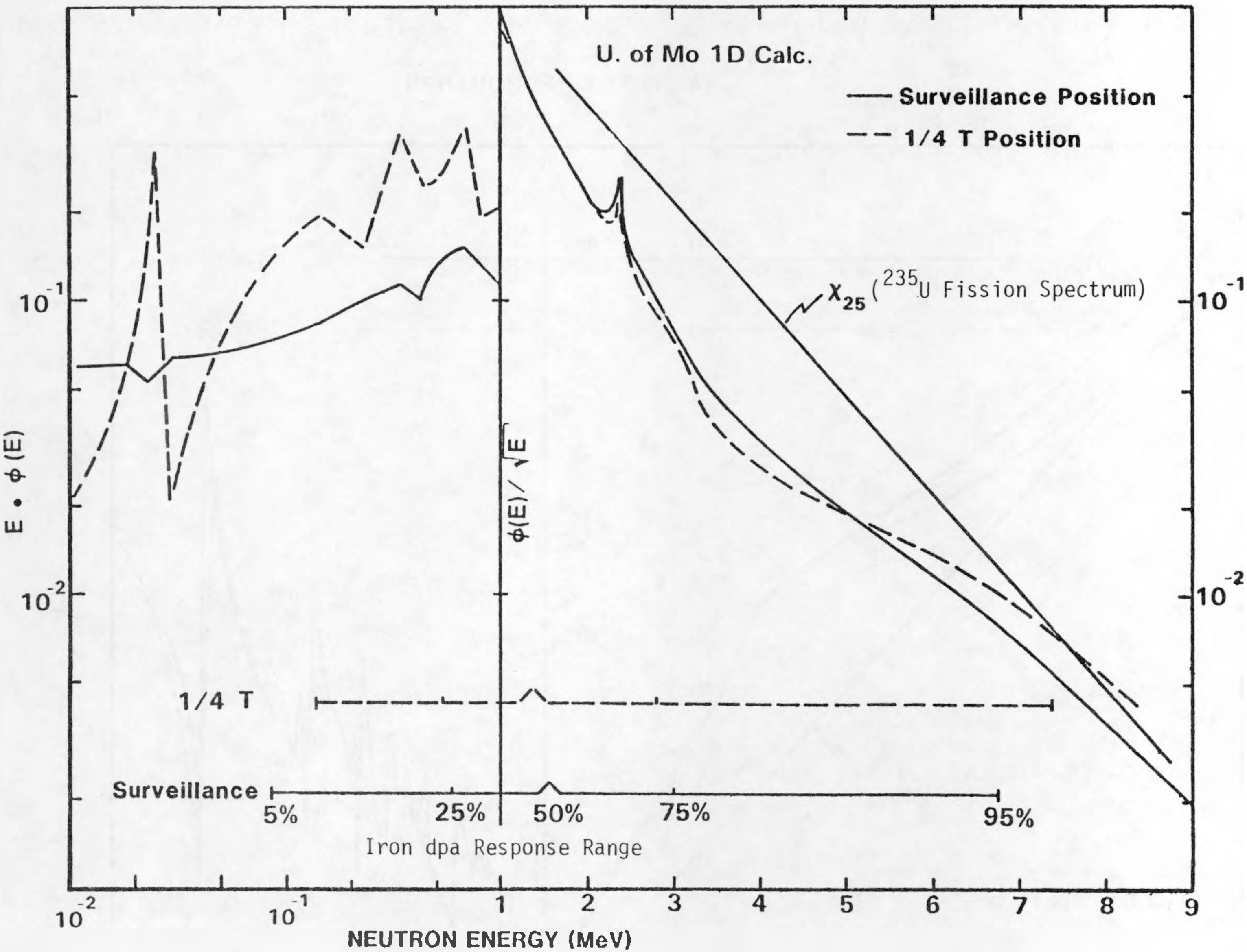


FIGURE 3.20. ANO-1 Surveillance and 1/4 T Position Neutron Spectra (taken from Reference 32).

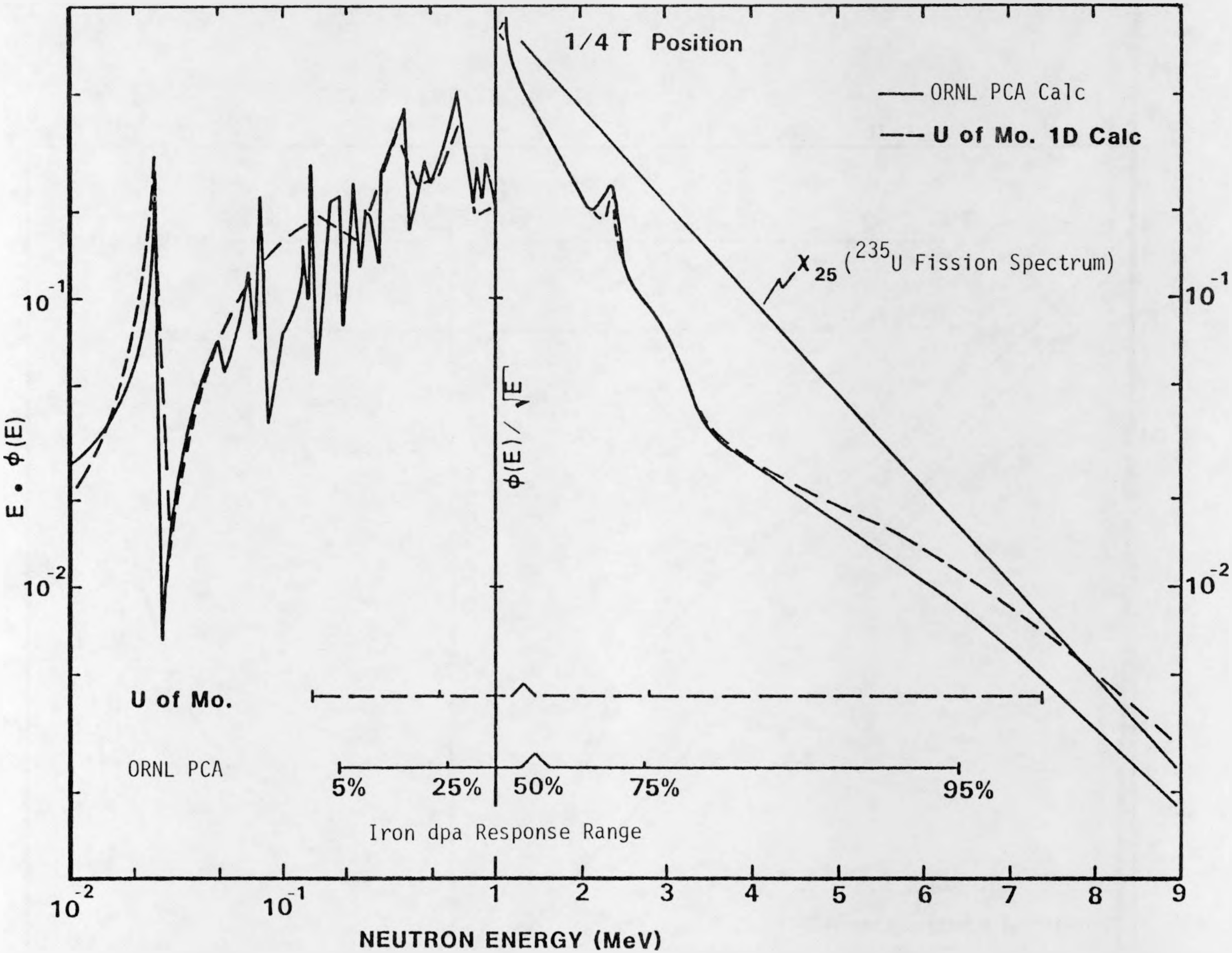


FIGURE 3.21. University of Missouri ANO-1 and ORNL PCA Calculation Comparisons (taken from Reference 32).

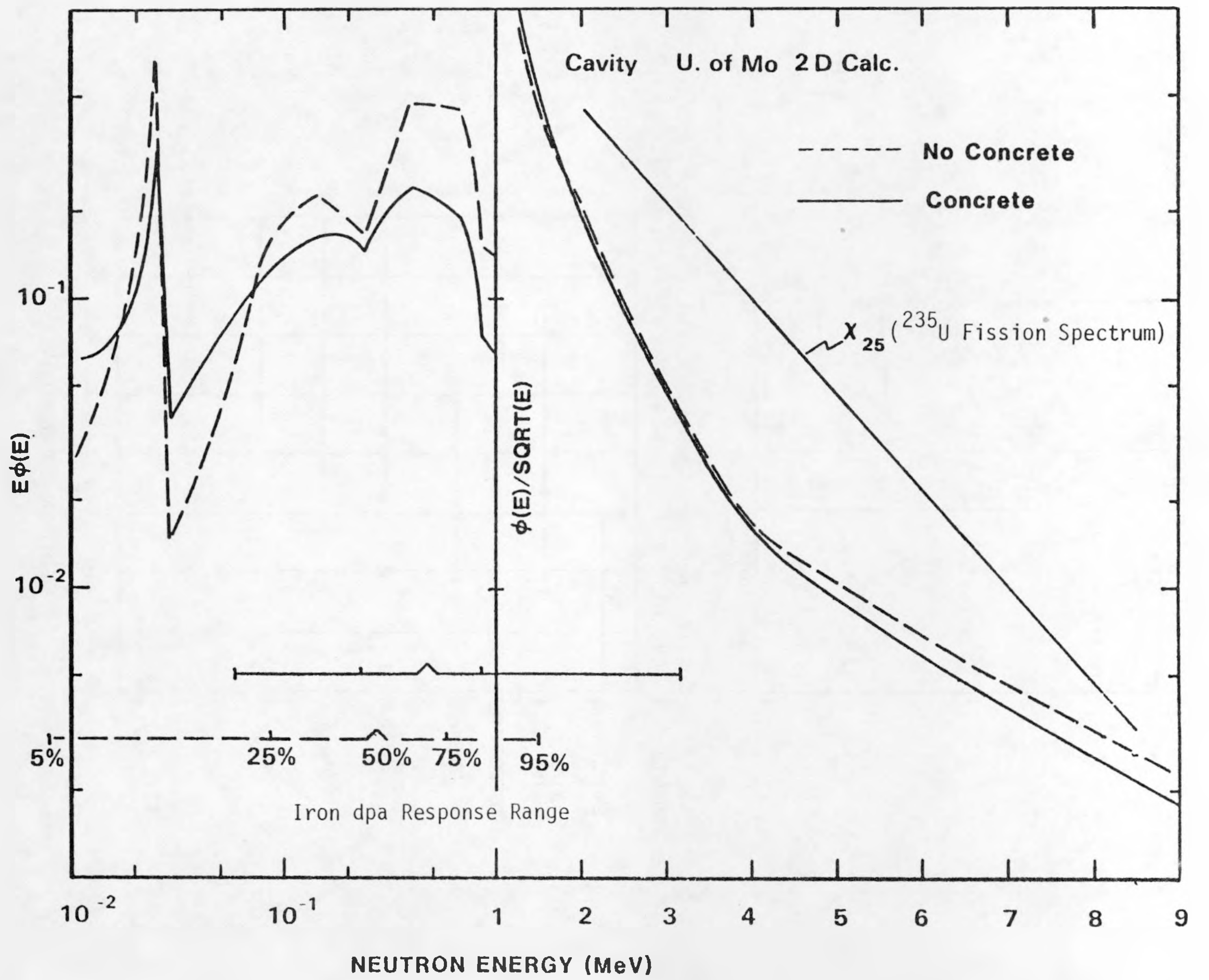


FIGURE 3.22. ANO-1 Cavity Neutron Spectra With and Without Concrete (taken from Reference 32).

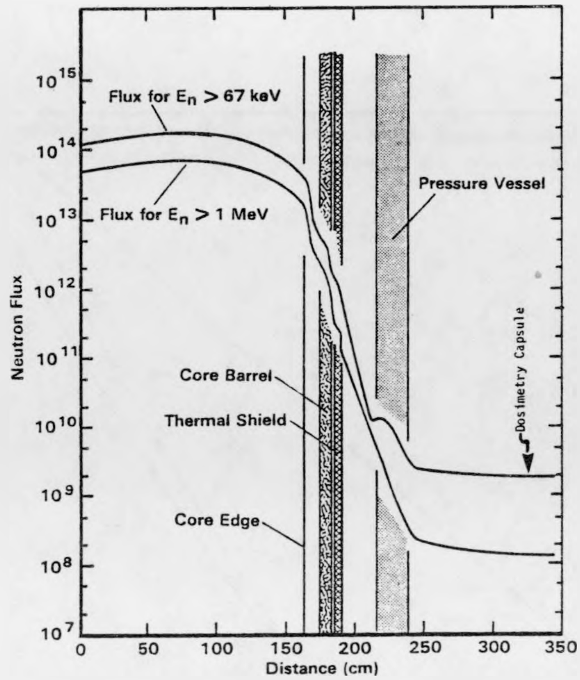


FIGURE 3.23. Flux Traverses for Two Spectrum Components, $\phi(E > 67\ \text{keV}, \text{ns}^{-1}\text{cm}^{-2})$ and $\phi(E > 1\ \text{MeV}, \text{ns}^{-1}\text{cm}^{-2})$ (taken from Reference 32).

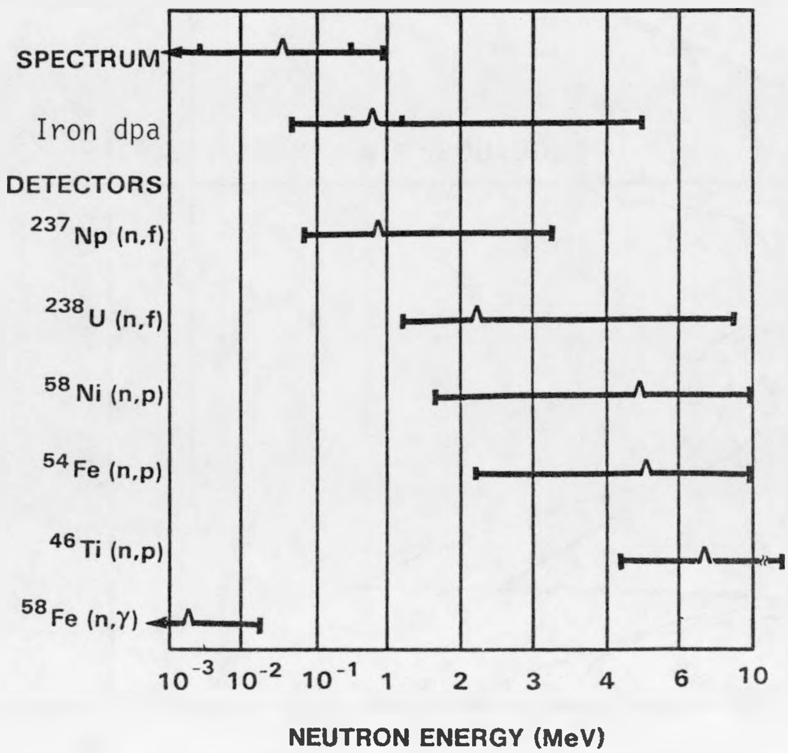


FIGURE 3.24. RM Detector Response Range ANO-1 Cavity Position (taken from Reference 32).

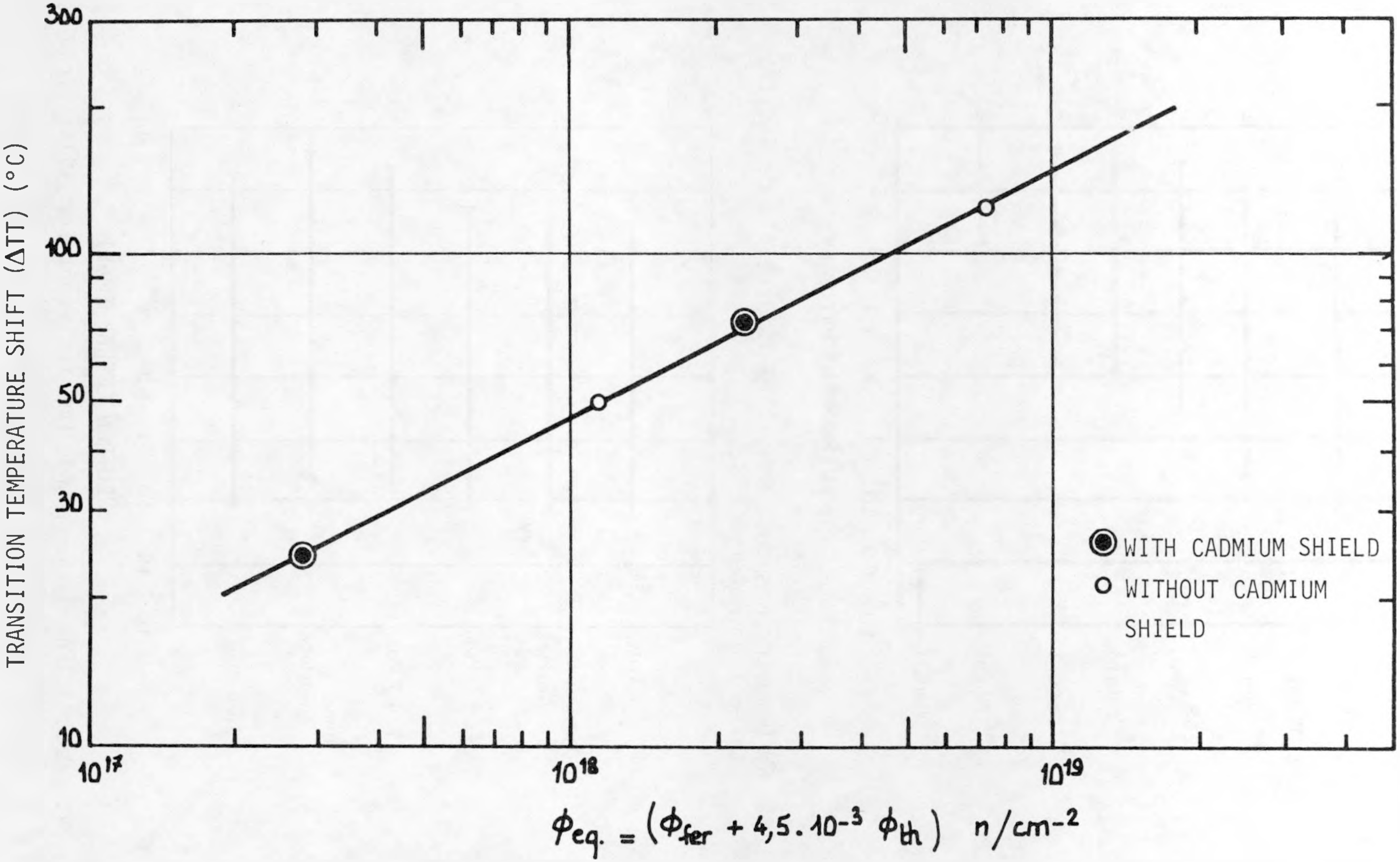


FIGURE 3.25. Influence of Fast and Thermal Neutron Exposures on the ΔTT Transition Temperature Shift for A537 Steel Irradiated at 60°C (taken from Reference 56).

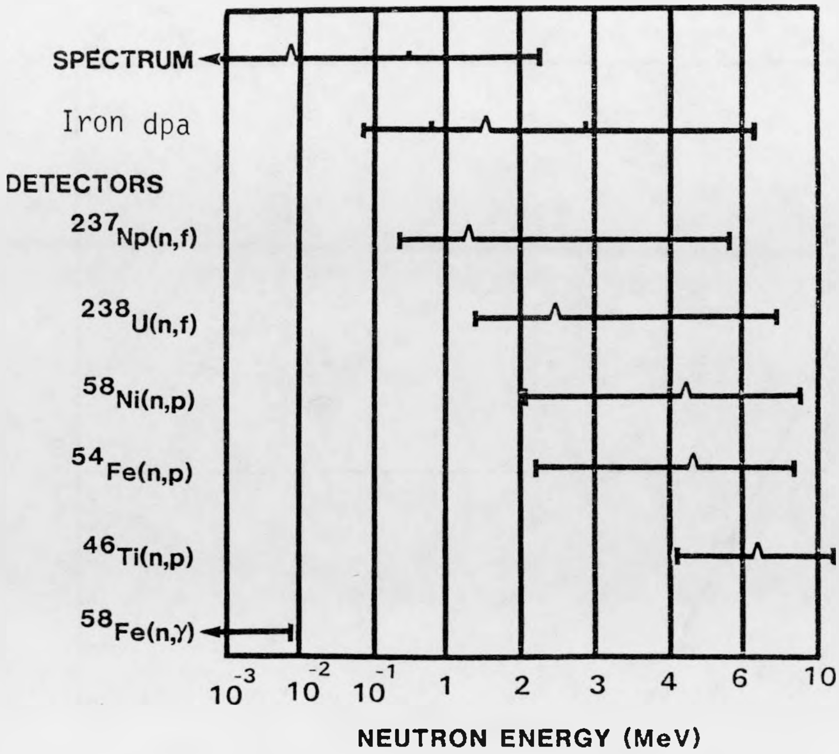


FIGURE 3.26. RM Detector Response Range ANO-1 Surveillance Position (taken from Reference 32).

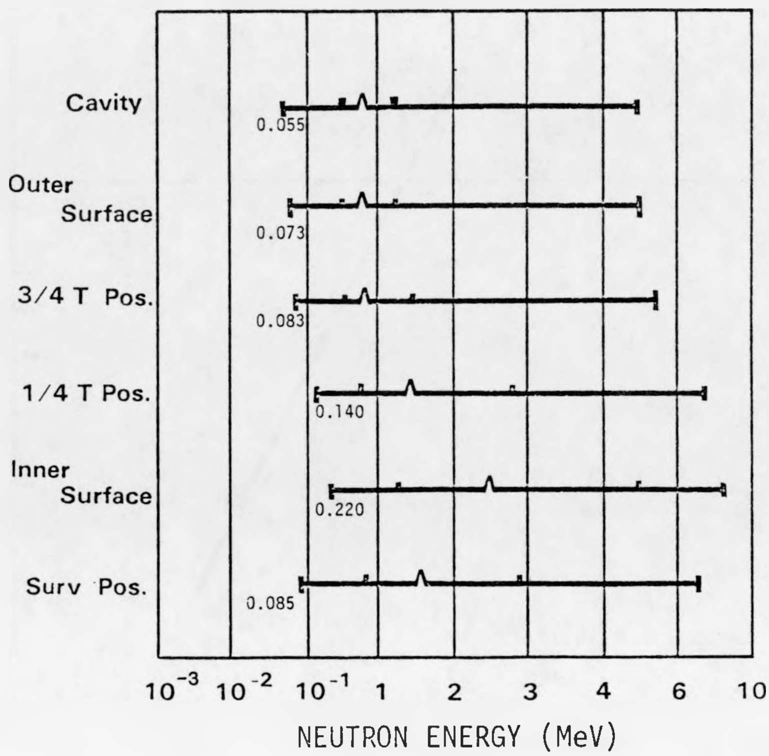


FIGURE 3.27. Neutron Energy Range for Embrittlement for ANO-1 (Iron dpa Exposure) (taken from Reference 32).

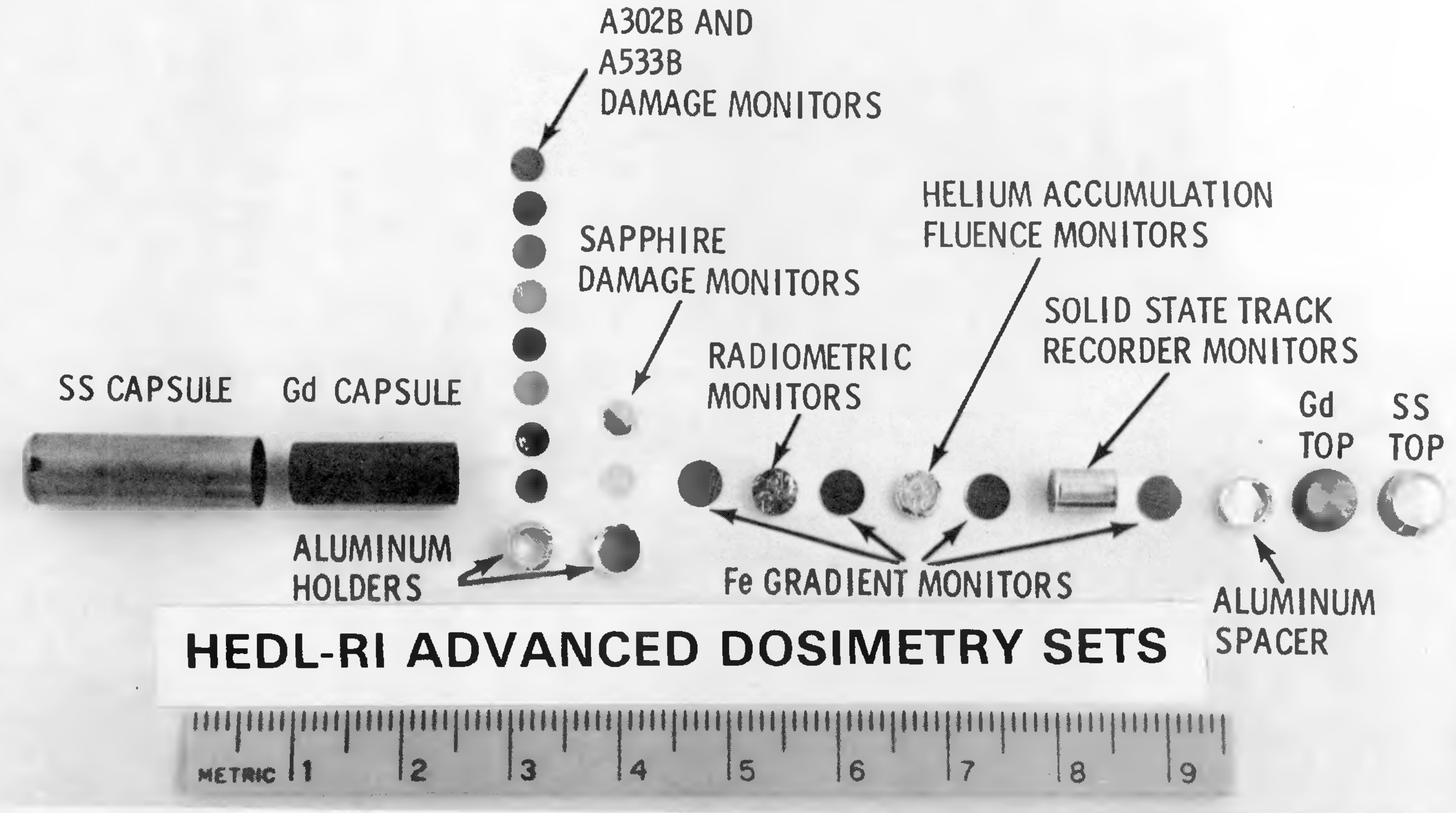
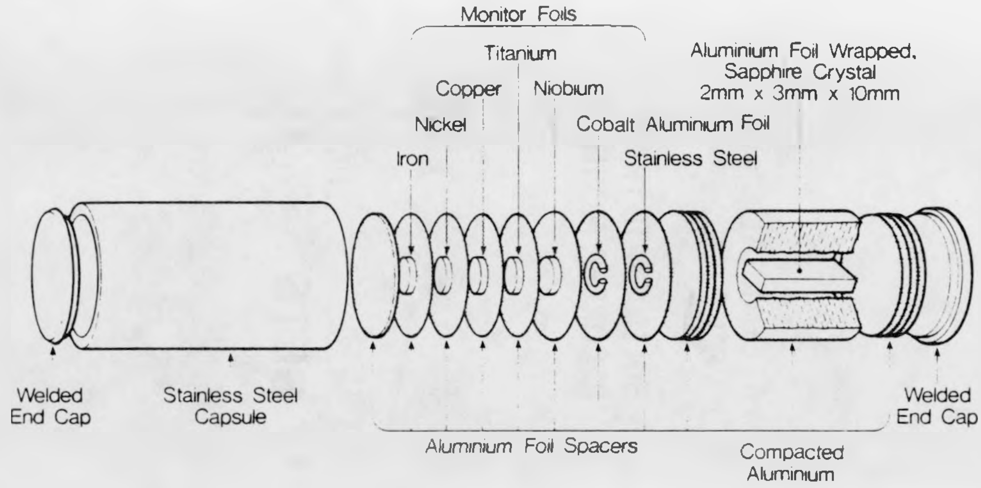
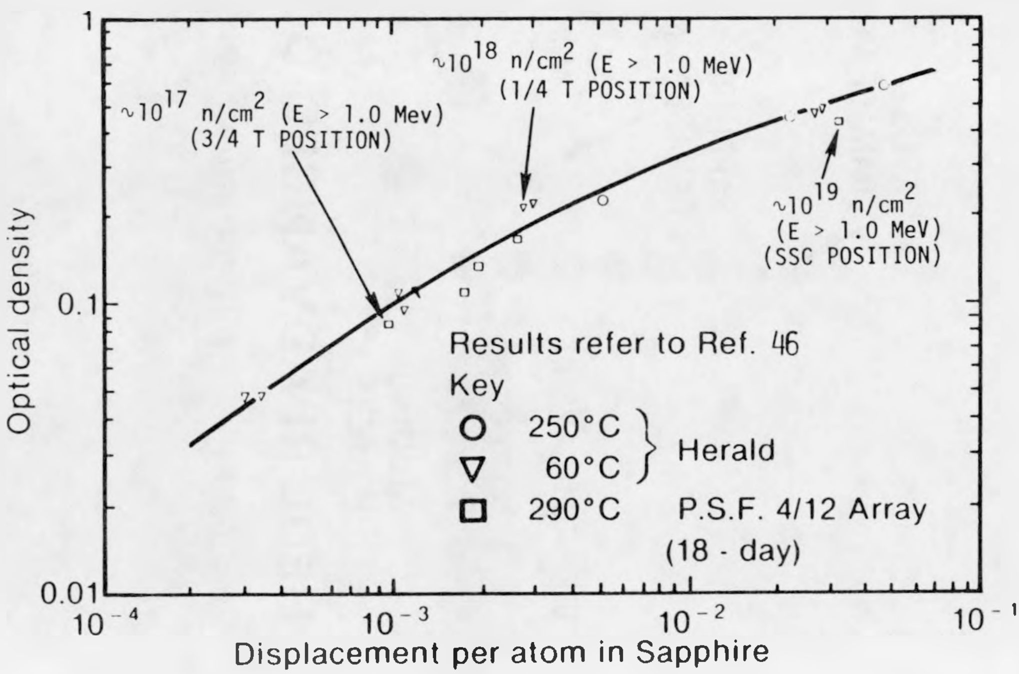


FIGURE 3.28. HEDL-RI Advanced Dosimetry Sets (taken from Reference 45). Neg 7909355-2



Radiometric and Sapphire Damage Monitor Irradiation Capsule for PSF Tests.



Irradiation Response of Sapphire Damage Monitors.

FIGURE 3.29. United Kingdom Advanced RM and DM Dosimetry Capsule and Sapphire Irradiation Damage Response (taken from Reference 29).

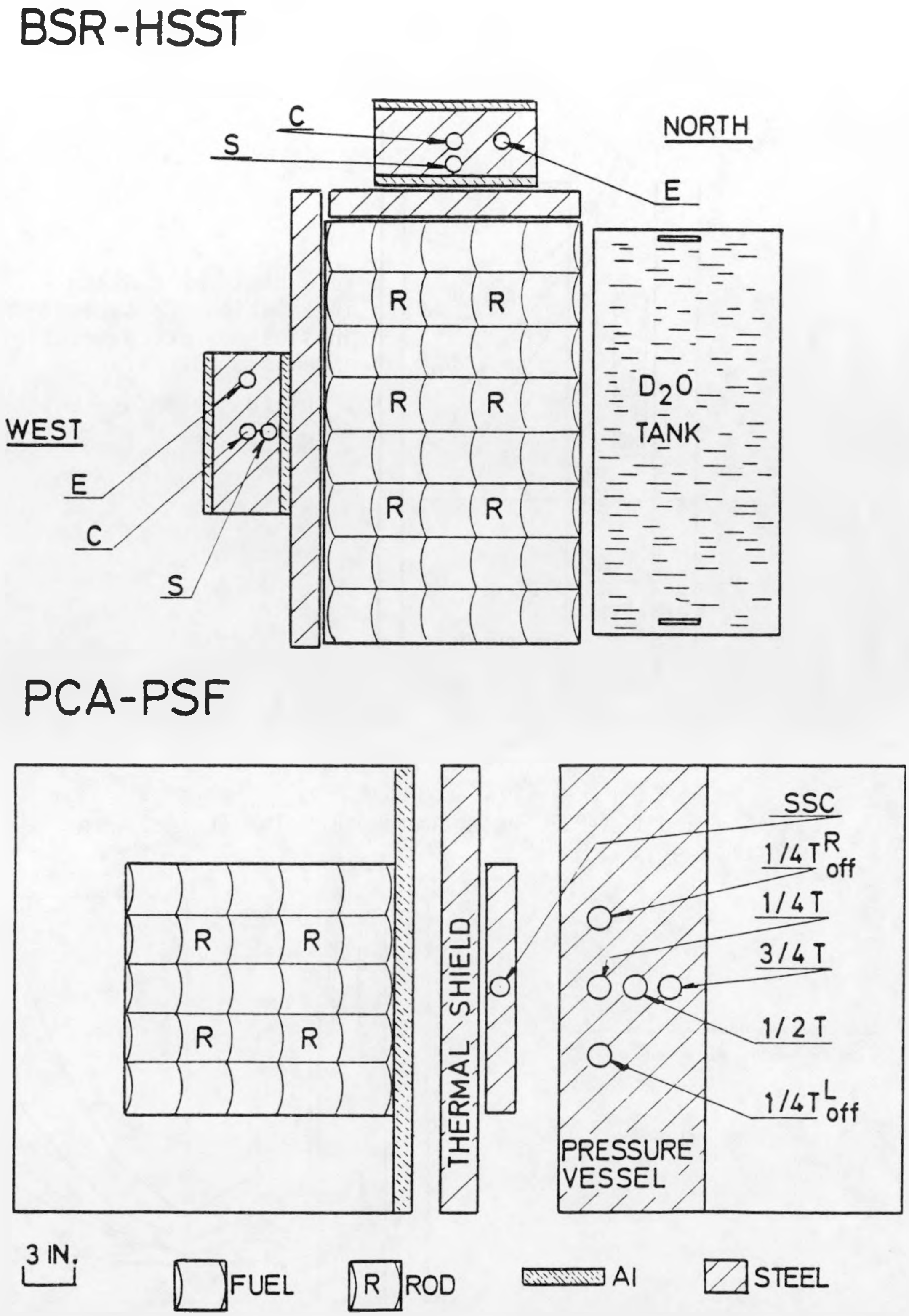
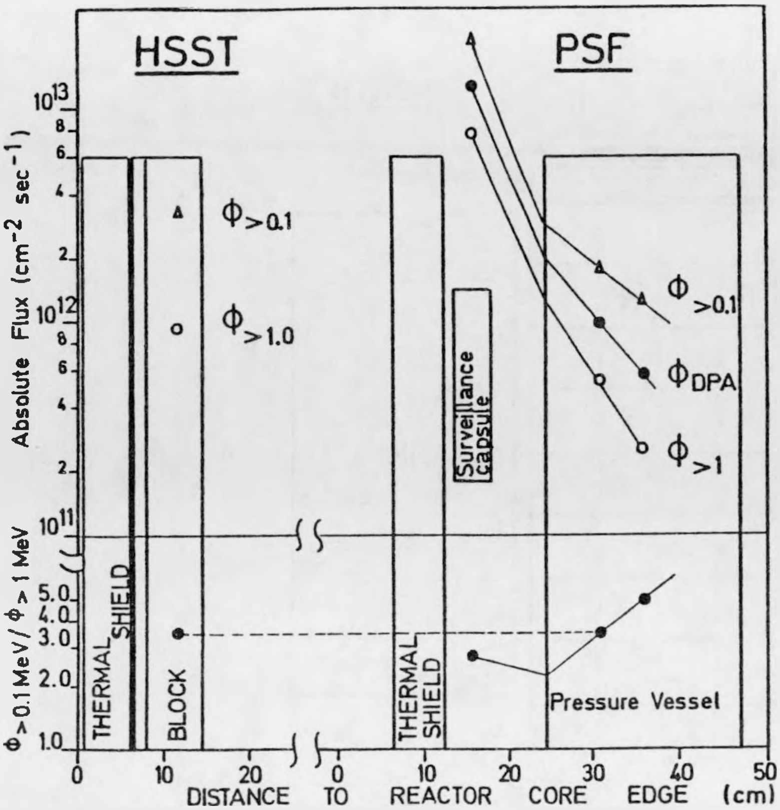


FIGURE 3.30. BSR-HSST and PCA-PSF Mockup Test Designs (taken from Reference 46).



Potentials of PSF and HSST irradiations to separate physics-dosimetry-metallurgy effects of different:

- Neutron exposure units
- Neutron flux level regimes

FIGURE 3.31. BSR-HSST and PCA-PSF Tests Summary Results (taken from Reference 46).

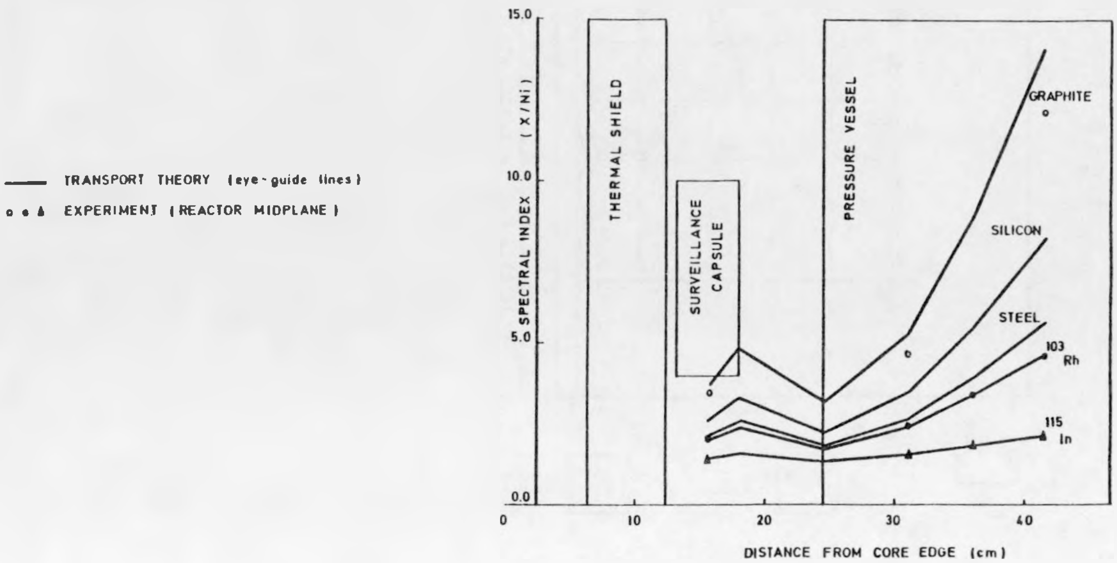


FIGURE 3.32. PSF Calculated and Measured Spectral Indices (taken from Reference 46).

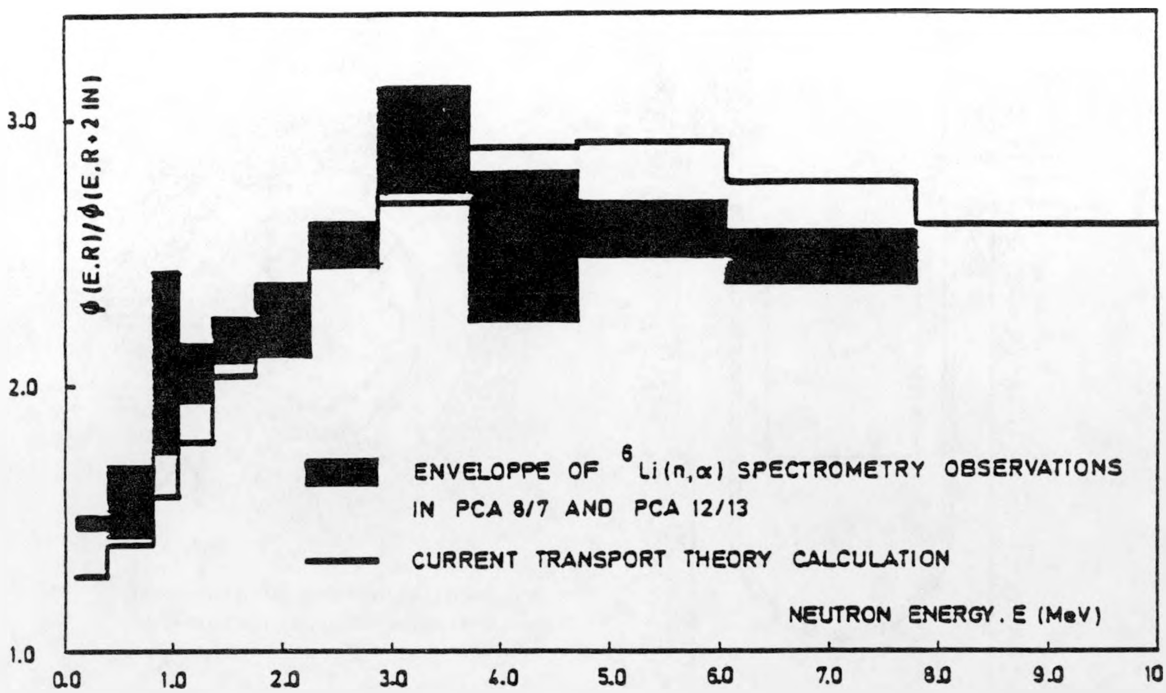


FIGURE 3.33. Radial Fission Flux Attentuation by Steel in Typical LWR Pressure Vessel Environments (taken from Reference 46).

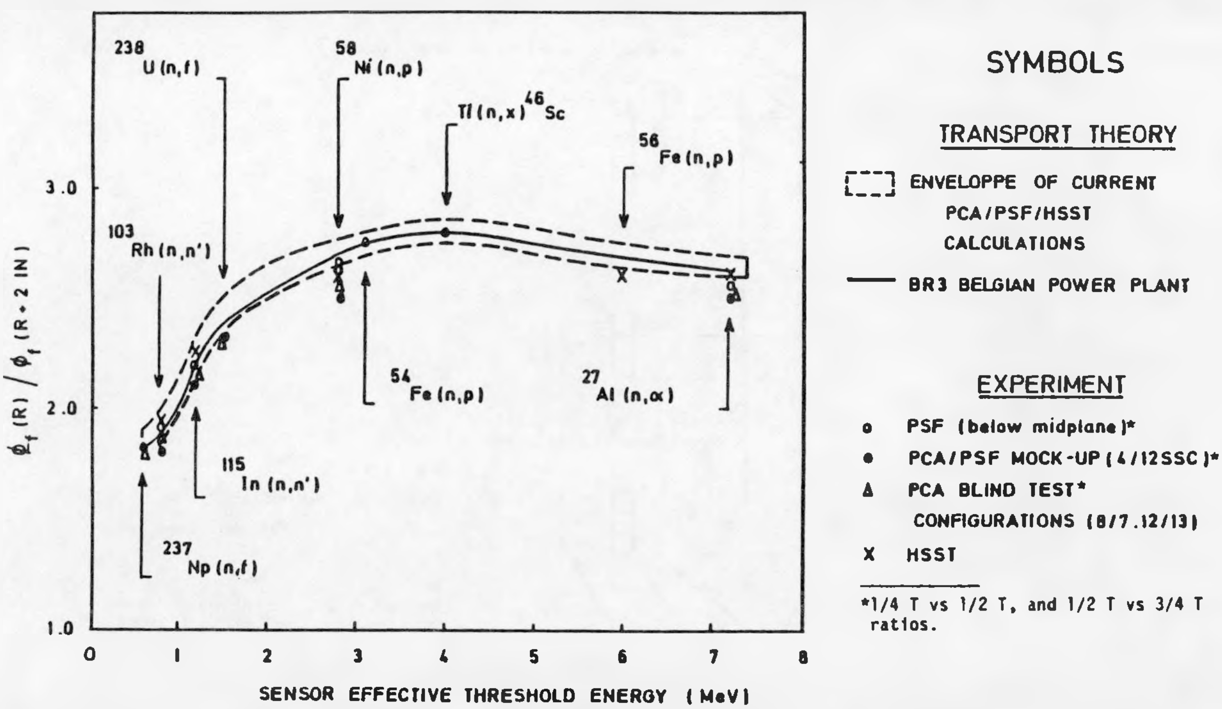


FIGURE 3.34. Radial Neutron Flux Spectrum Attenuation by Steel in Simulated LWR Pressure Vessel Environments (taken from Reference 46).

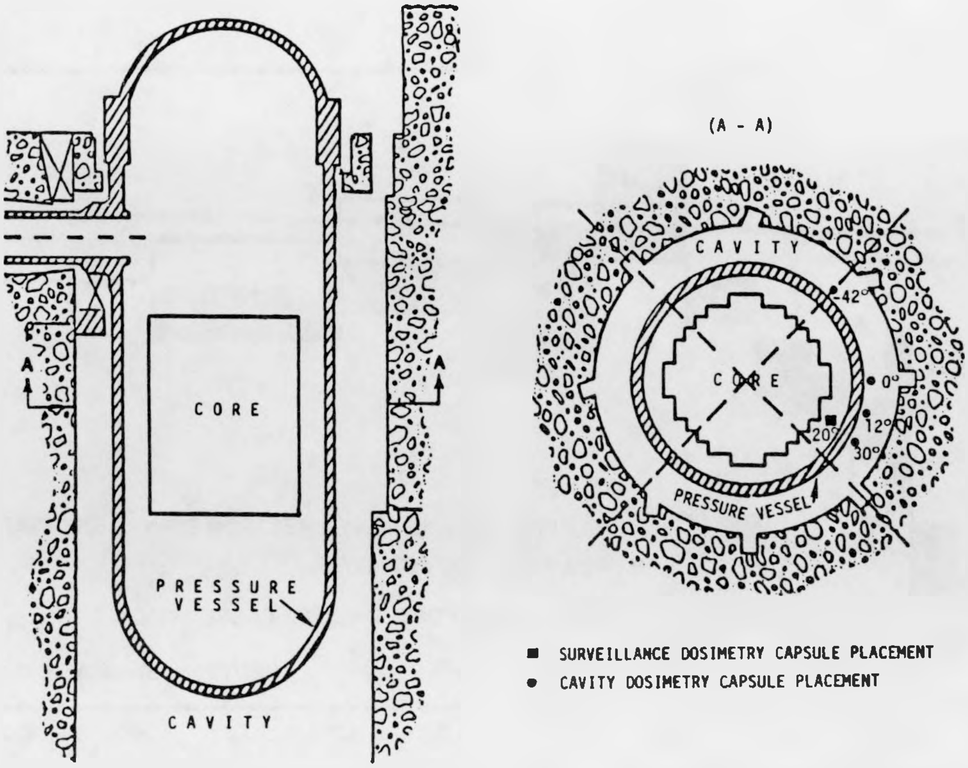


FIGURE 3.35. Typical 3-Loop Westinghouse PWR: Schematic Representation for H. B. Robinson Surveillance and Cavity Dosimetry Capsule Placement.

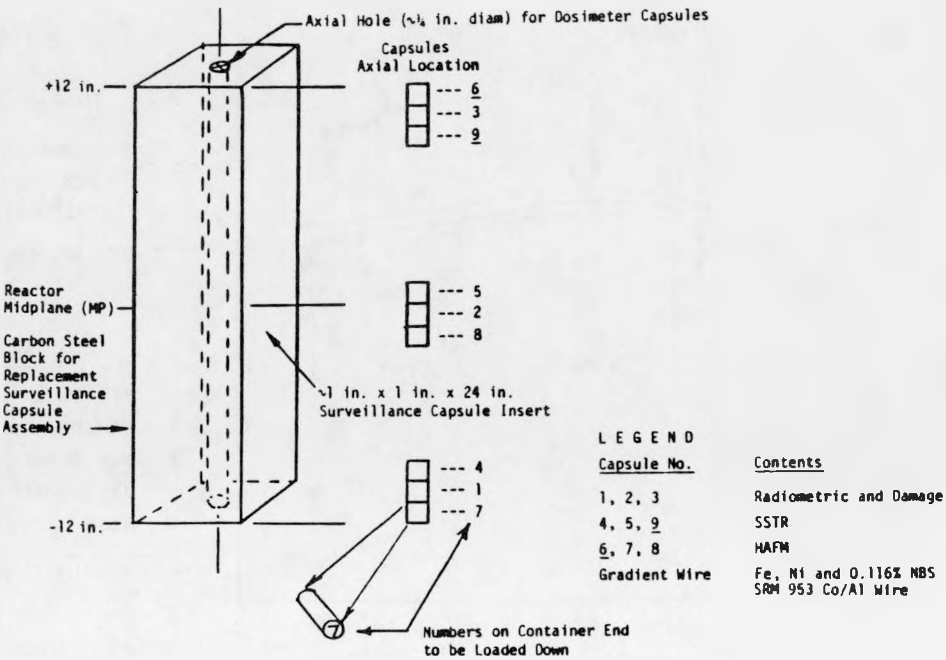


FIGURE 3.36. H. B. Robinson Surveillance Capsule Dosimetry.

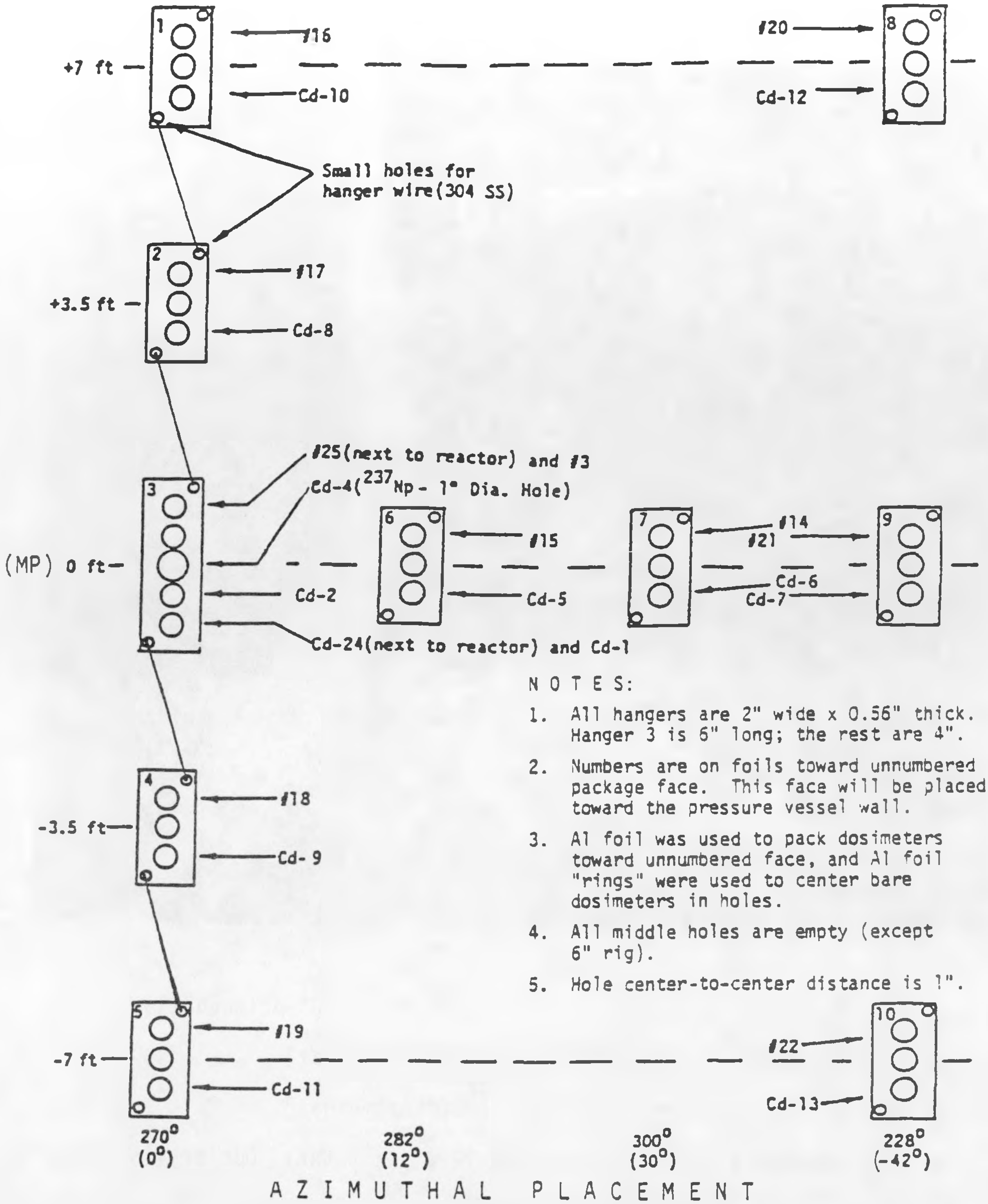


FIGURE 3.37. H. B. Robinson Cavity Dosimetry Hanger Rigs. (Individual dosimeter sensors are identified in Table 4).

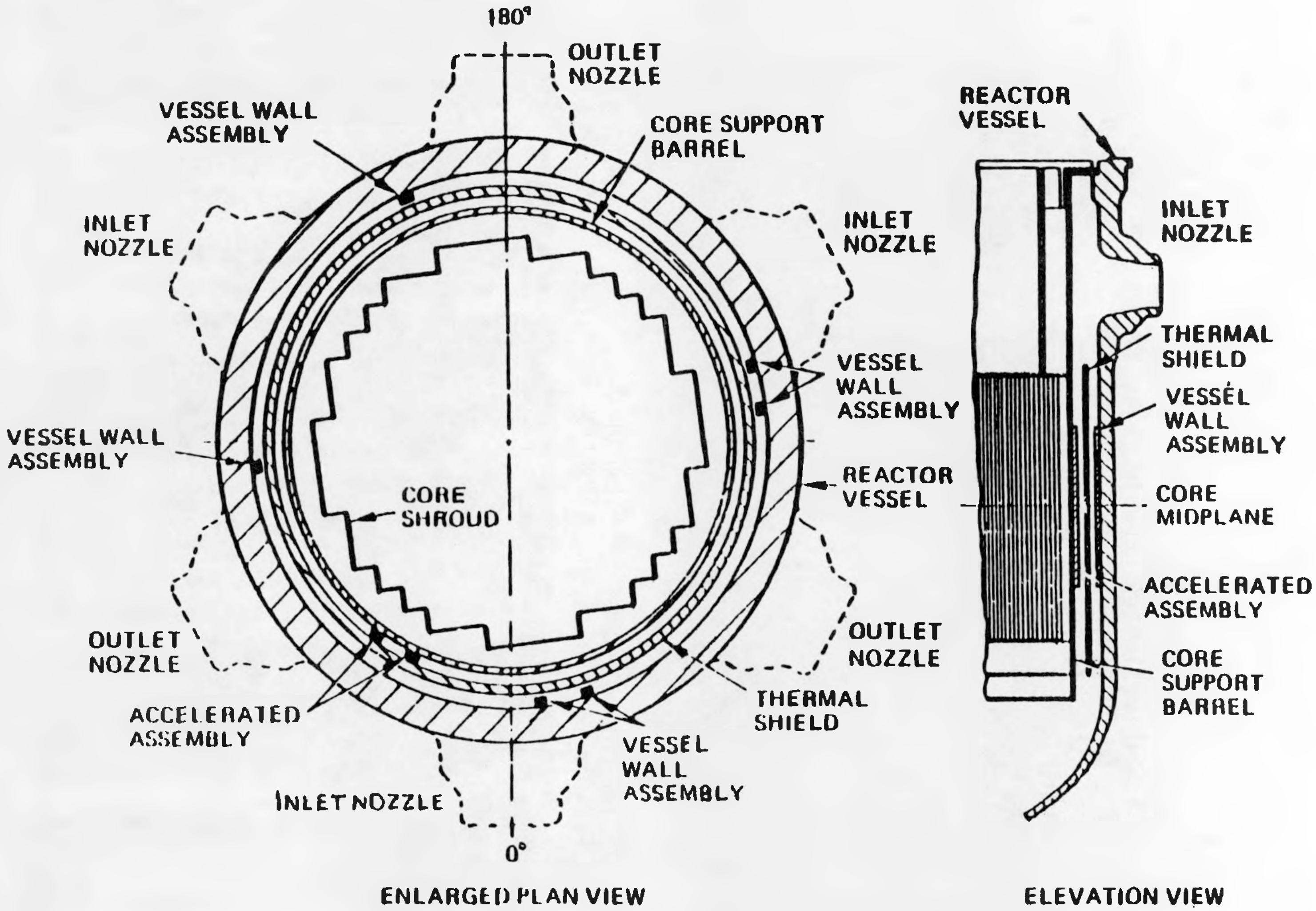
1) 0° Dosimetry String (1 SSTR & 5 RM Sets) [270° Azimuthal]

2) 12° Dosimetry String (1 RM Set) [282° Azimuthal]

3) 30° Dosimetry String (1 RM Set) [300° Azimuthal]

4) -42° Dosimetry String (3 RM Sets) [228° Azimuthal] Out of View

FIGURE 3.38. Actual Placement of Cavity Dosimetry Hanger Rigs for H. B. Robinson. Neg 8205833-2cn



18534-13

FIGURE 3.39. Typical Locations of Maine Yankee Surveillance Capsule Assemblies. (The three selected cavity locations are not shown, and actual placement has yet to be accomplished.)

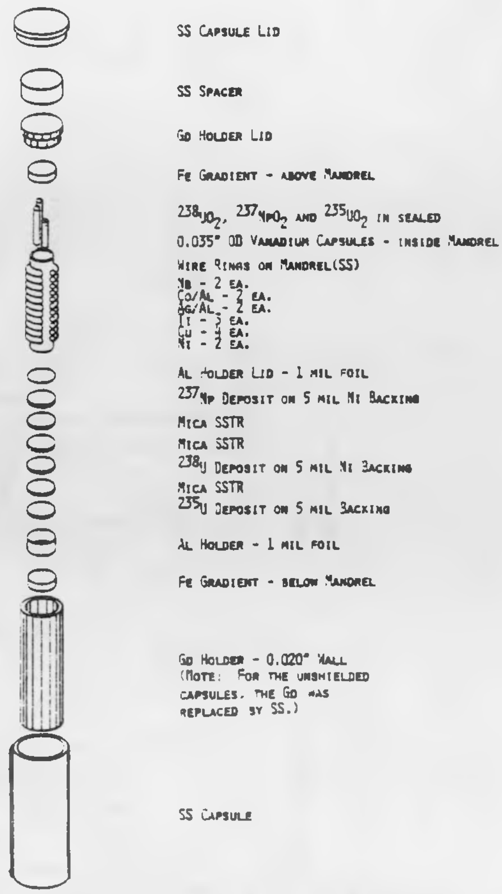


FIGURE 3.40. Maine Yankee Midplane RM and SSTR Dosimetry Capsule with Gd Shield. (There are 3 Gd-shielded and 3 unshielded capsules for the top, midplane and bottom locations.)

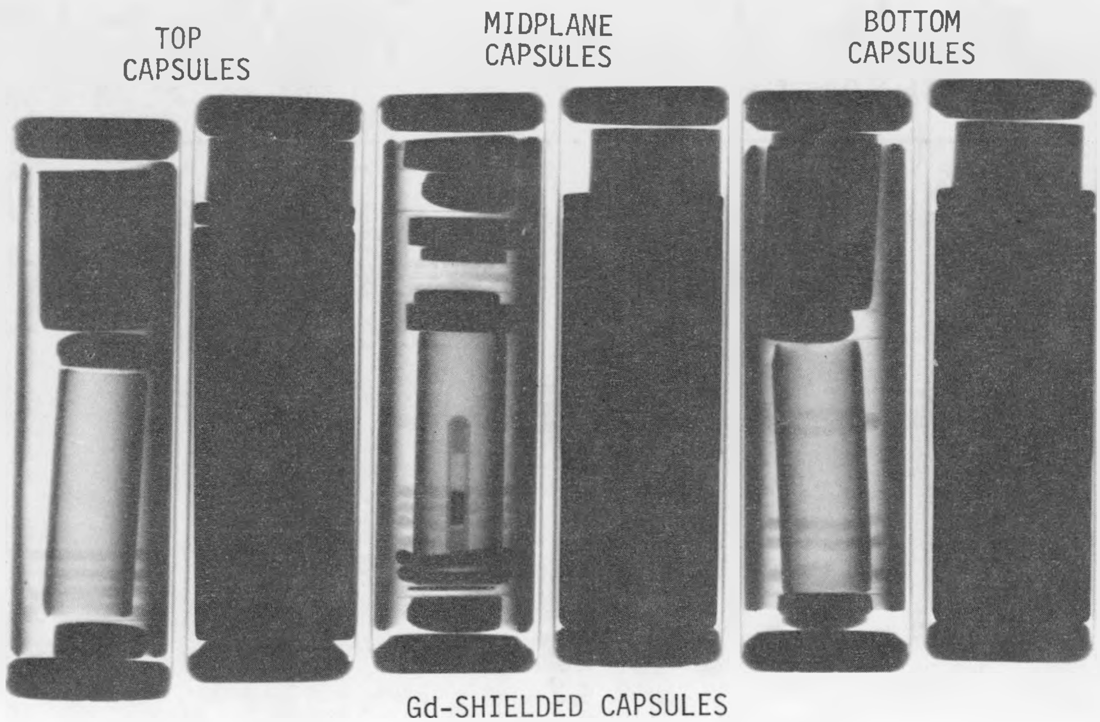


FIGURE 3.41. Maine Yankee Surveillance Capsules: Quality Assurance Radiographs for Capsule Weld Integrity and Sensor Placement Verification.

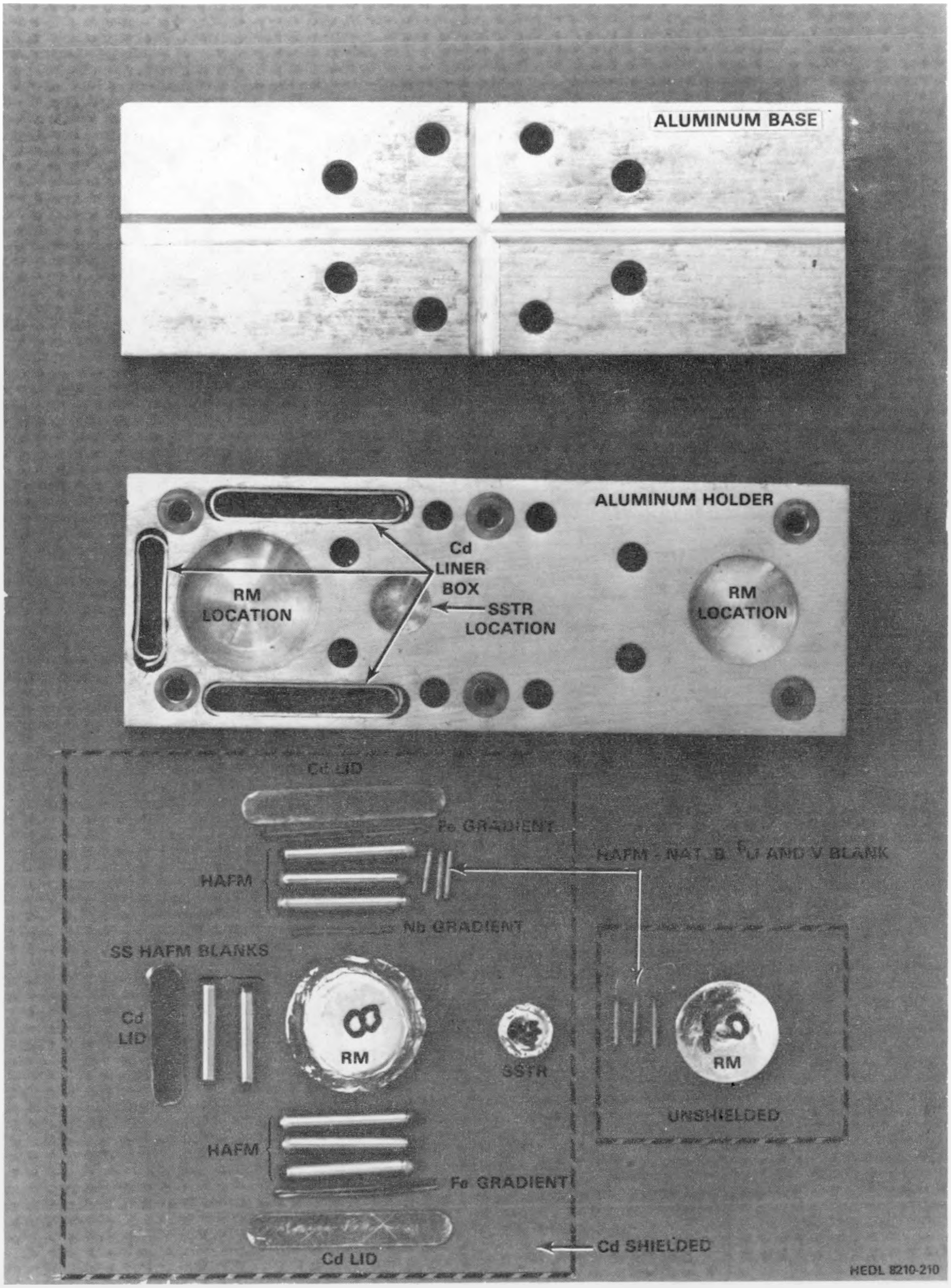


FIGURE 3.42. Maine Yankee 15° and 30° Cavity Dosimetry Holder with RM, SSTR, HAFM and Gradient Wires Before Assembly.

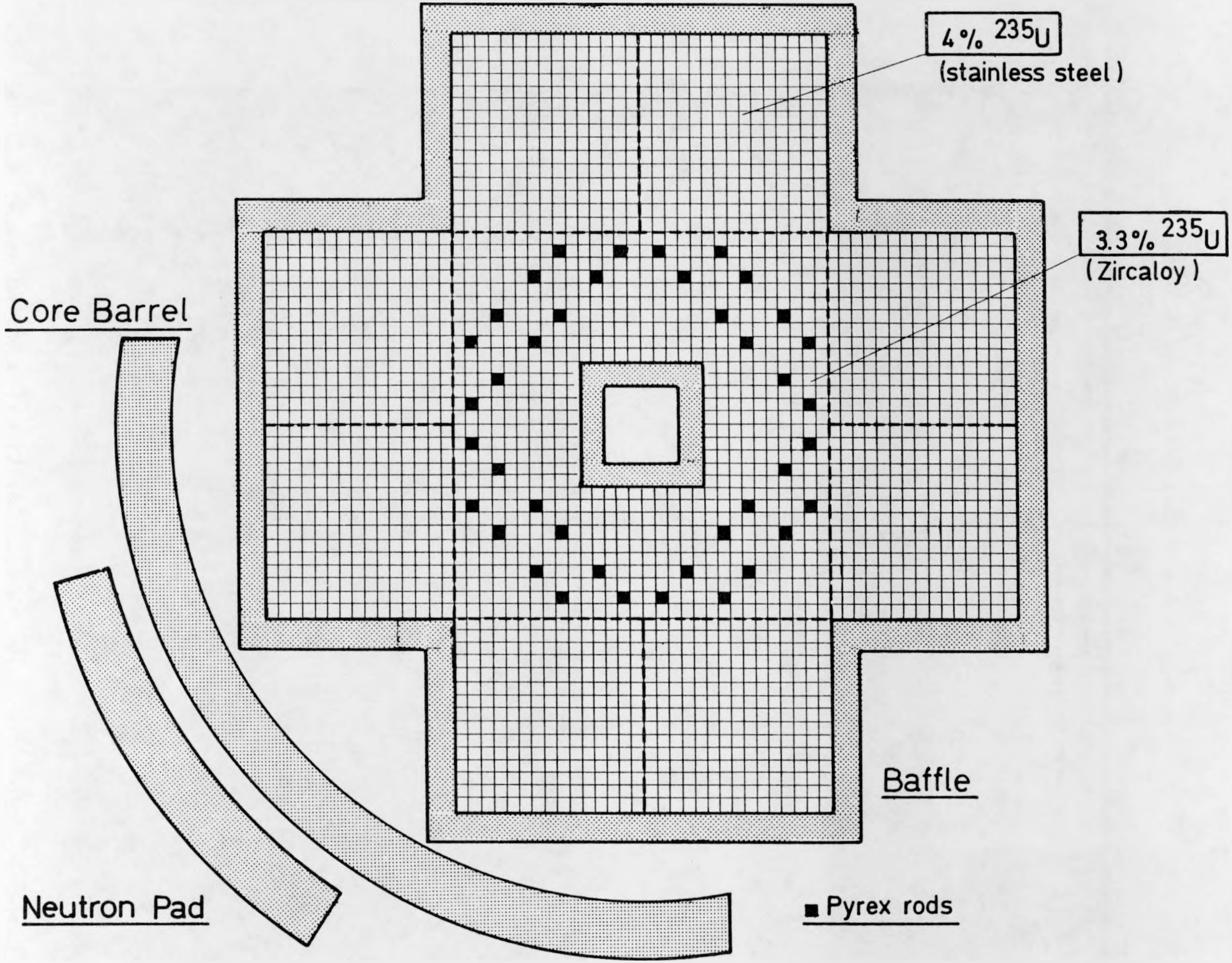


FIGURE 3.43. VENUS Benchmark (taken from Reference 60).

NESDIP

NESTOR DOSIMETRY IMPROVEMENT PROGRAMME

OBJECTIVES

1. To provide a 'clean-source' UK PV-Steels benchmark experiment for methods-testing.
2. To extend scope of US-NRC/SDIP benchmark programme in important areas of interest.
3. To complement information from other international dosimetry programmes.

Fig. 1

SCOPE OF **NESDIP** PROGRAMME

PHASE 1	ORNL-PCA 'REPLICA' (Neutronics checks; PSF methods checks; extended γ ray measurements)
PHASE 2	SIMULATED PV - CAVITY (Development of Cavity-monitoring and interpolation; Cavity size effects; neutron streaming corrections)
PHASE 3	SIMULATED PV - SUPPORT STRUCTURE (PV - nozzle effects; support structure dosimetry)

Fig. 2

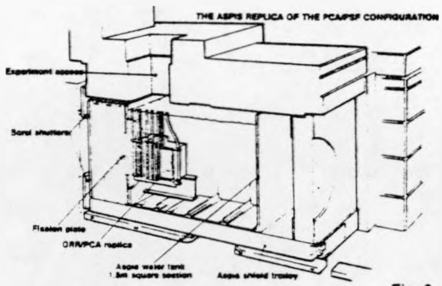


Fig. 3

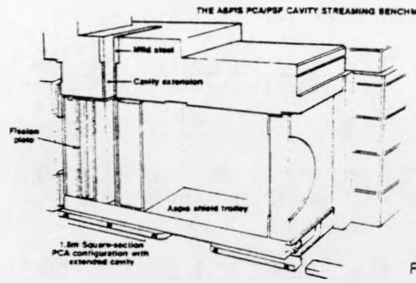


Fig. 4

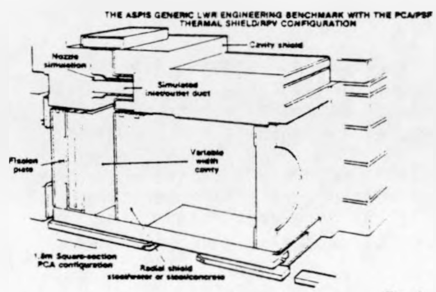


Fig. 5

NESDIP

PROPOSED MEASUREMENT TECHNIQUES

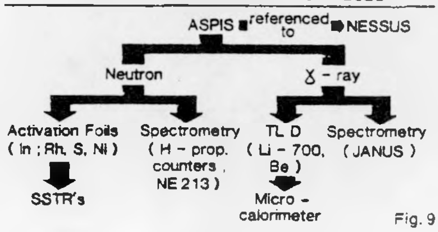


Fig. 9

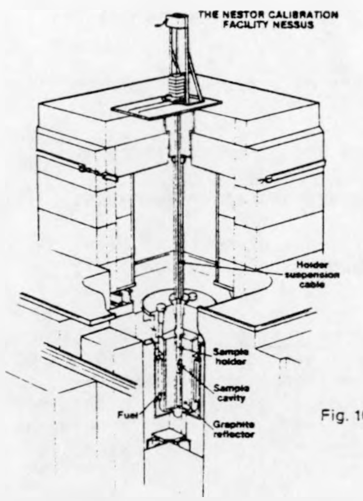


Fig. 10

NESDIP

DRAFT PROGRAMME PROPOSALS

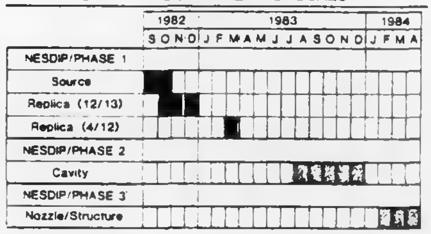


Fig. 11

NESDIP

PROGRAMME CONTEXT

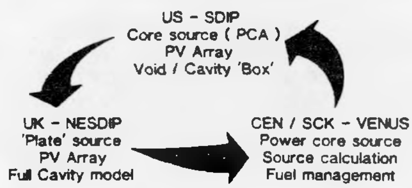


Fig. 12

FIGURE 3.44. NESDIP Benchmark (taken from Reference 61).

TABLE 3.1

LICENSING AND REGULATORY REQUIREMENTS RELATED TO THE ASSESSMENT AND CONTROL OF THE FRACTURE TOUGHNESS OF REACTOR PRESSURE VESSELS

A. Two distinct licensing requirements form the backbone of the latest regulations related to the fracture toughness of reactor pressure vessels:*

1. Protection against failure by tearing instability:
(Ductile regime, 100% shear fracture)

- USE $\geq$ 50 ft-lb (67.8 joules) (1)

(USE is the Upper Shelf Energy absorbed in the C_v-impact test at the vessel operating temperature)

2. Protection against non-ductile failure:

Applied Load x Safety Margin $\leq$ Material Strength (2)

$$2\ K_{IP} + K_{IT} \leq K_{IR}\ (T\text{-}RT_{NDT})$$

Pressure + Thermal + + (Calculated Stress Intensity Factors)	Reference Fracture Toughness K_{IR} = Lower Bound of Valid K_{Ic}, K_{Ia}, K_{Id} Measurements (Indexed to reference temperature, $T\text{-}RT_{NDT}$)[7]

where RT_{NDT} = (unirradiated nil-ductility temperature) + (ΔRT_{NDT}). From this relationship are derived the pressure versus temperature heat-up and cool-down limit curves P(T): at core criticality, these limits must, furthermore, be shifted conservatively by an additional margin of 40°F.

B. Surveillance-capsule physics-dosimetry measurement results enter into the application of requirements of Eq. (1) and (2) at two stages:

1. Mechanical testing and physics-dosimetry data are used to consolidate plant-specific "trend curves":

USE = function of neutron exposure and other variables (3)

ΔRT_{NDT} = function of neutron exposure and other variables (4)

The neutron exposure is expressed as fluence of neutrons with energy greater than 1 MeV or, more appropriately, as dpa.[30]**

2. Dosimetry data are used to consolidate reactor physics calculations of in-vessel neutron exposure projections (lead factors) at the end of the considered plant service cycle: The derived exposures are then input to Eqs. (3) and (4) in order to apply Eqs. (1) and (2); in this regard, ex-vessel dosimetry measurements[32-38] are a particularly relevant supplement to surveillance capsule dosimetry and to the extensive low power benchmarking studies in PCA,[23] VENUS[18] and NESDIP.[29]**

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*In addition, screening criteria to sort out plants for which more extensive analysis of thermal shock risk is needed have recently been proposed by the NRC.

**Physics-dosimetry licensing requirements are as yet unspecified, but the technology and the ASTM Standards are at hand for the use of dpa and ex-vessel measurements, see Refs. 17, 30 and 39.

TABLE 3.2*

PROCEDURES FOR ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS

PROCEDURAL STEPS:

1. Establish the basic surveillance test program for each operating power plant. Currently Practice E185 is available and is used. However, updated versions of this standard should include the following:

2. Determination of surveillance capsule spatial flux-fluence-spectral and DPA maps for improved correlation and application of measured property change data (upper shelf, NDTT, etc.). Measured surveillance capsule fission and nonfission monitor reaction and reaction rate data should be combined with reactor physics computations to make necessary adjustments for capsule perturbation effects.

3. As appropriate, use of measured/calculated DPA damge for normalization of Charpy to Charpy (and other metallurgical specimen) variations in neutron flux, fluence, and spectra. Here, an increased use of a larger number of metallurgical specimen iron drillings may be appropriate for dosimetry.

4. Establish a reactor physics computational method applicable to the surveillance program. Currently Practices E 482 and E 560 provide general guidance in this area. However, updated versions of these standards should include the following:

5. Determination of core power distributions applicable to long-term (30 to 40 year) irradiation. Associated with this is the need for the use of updated FSAR (Final Safety Analysis Report) reactor physics information at startup.

6. Determination of potential cycle-to-cycle variations in the core power distributions. This will establish bounds on expected differences between surveillance measurements and design calculations. Ex-vessel dosimetry measurements should be used for verification of this and the previous step.

7. Determination of the effect of surveillance capsule perturbations and photofission on the evaluation of capsule dosimetry. Adjustment codes should be used, as appropriate, to combine reactor physics computations with dosimetry measurements.

8. Benchmark validation of the analytical method.

9. Establish methods for relating dosimetry, metallurgy, and temperature data from the surveillance program to current and future reactor vessel and support structure conditions. Currently, Practice E 560 provides general guidance in this area. An updated version of this standard should include the following considerations:

10. Differences in core power distributions that may be expected during long-term operation and that may impact the extrapolation of surveillance results into the future. As previously stated, ex-vessel dosimetry should be used for verification.

11. Establish methods to verify Steps 2 - 10 and to determine uncertainty and error bounds for the interpretation of the combined results of dosimetry, metallurgical and temperature measurements. Currently, Practice E185 provides general guidance in this area. An updated version of this standard should more completely address the separate and combined accuracy requirements of physics, dosimetry, metallurgy, and temperature-measurement techniques.

*Taken from ASTM Standard E 853-81.(39)

TABLE 3.3

CURRENT ANALYSIS PROCEDURES AND DATA USED BY A NUMBER OF US LABORATORIES AND VENDORS

Analyst	Transport Code Used	Transport Code Cross-Section Data	Sensor Cross-Section Data	Adjustment Code	Currently Reported Exposure Values
Westinghouse	DOT IIIW	ENDF/B-II, -III & -IV adjusted in-house	ENDF/B-IV	SACSBOT*	E > 1.0 MeV Fluence Thermal Fluence dpa
General Electric	DOT II Variant (SN2D)	ENDF/B-IV	ENDF/B-V	GE-RD-M02	E > 1.0 MeV Fluence E > 0.1 MeV Fluence Thermal Fluence Some use of dpa
Combustion Engineering	DOT III Changing to IV.2	DLC-23E (Cask)	SAND-II Library	SAND-II	E > 1.0 MeV Fluence Thermal Fluence dpa
Babcock & Wilcox	Previously DOT III.5 now IV.2	DLC-23E (Cask)	ENDF/B-V	Equivalent to SACSBOT*	E > 1.0 MeV Fluence E > 0.1 MeV Fluence Thermal Fluence
Brookhaven	DOT III.5	ENDF/B-IV	Collapsed Version of ENDF/B-V	SACSBOT*	E > 1.0 MeV Fluence E > 0.1 MeV Fluence Thermal Fluence dpa
SWRI	DOT III.5 Changing to IV.2	DLC-23E (Cask) Changing to DCL-75 BUGLE-80 (ENDF/B-IV)	ENDF/B-IV Changing to ENDF/B-V	Previously SAND-II now SACSBOT*	E > 1.0 MeV Fluence E > 0.1 MeV Fluence Thermal Fluence
BMI	DOT III.5 Changing to IV.2	DLC-23E (Cask) Changing to DCL-75 BUGLE-80 (ENDF/B-IV)	SAND-II Library	SACSBOT*	E > 1.0 MeV Fluence E > 0.1 MeV Fluence Thermal Fluence dpa

*SACSBOT = Individual Sensor Spectrum Averaged Cross Sections Based On Transport Calculations.

TABLE 3.4

REEVALUATED EXPOSURE VALUES AND THEIR UNCERTAINTY FOR LWR PRESSURE VESSEL SURVEILLANCE CAPSULES (Revision of Reference 40 data)

Plant	Unit	Capsule	Fluence (ϕt > 1 MeV) (n/cm^2) Old	New [% (1σ)]	New/Old	dpa [% (1σ)]	dpa/ϕt New	dpa/s	Exposure* Time (s)
Westinghouse									
Conn. Yankee		A	2.08 + 18	3.17 + 18 (12)**	1.52	4.89-03 (12)	1.54-21	9.18-11	5.233 + 07
Conn. Yankee		F	4.04 + 18	6.17 + 18 (24)	1.53	9.70-03 (27)	1.57-21	1.27-10	7.651 + 07
Conn. Yankee		H	1.79 + 19	2.06 + 19 (25)	1.15	3.38-02 (28)	1.64-21	1.42-10	2.390 + 08
San Onofre		A	1.20 + 19	2.93 + 19 (22)	2.44	5.04-02 (27)	1.72-21	8.66-10	5.824 + 07
San Onofre		D	2.36 + 19	5.66 + 19 (26)	2.40	9.51-02 (29)	1.68-21	1.07-09	8.881 + 07
San Onofre		F	5.14 + 19	5.81 + 19 (14)	1.13	9.79-02 (21)	1.69-21	4·.02-10	2.438 + 08
Turkey Pt.	3	S	1.41 + 19	1.66 + 19 (25)	1.18	2.65-02 (27)	1.60-21	2.42-10	1.095 + 08
Turkey Pt.	3	T	5.68 + 18	7.05 + 18 (10)	1.24	1.09-02 (12)	1.55-21	4.74-10	2.302 + 07
Turkey Pt.	4	S	1.25 + 19	1.34 + 19 (25)	1.07	2.22-02 (27)	1.66-21	2.06-10	1.079 + 08
Turkey Pt.	4	T	6.05 + 18	7.58 + 18 (13)	1.25	1.32-02 (13)	1.74-21	3.53-10	3.728 + 07
H. B. Robinson	2	S	3.02 + 18	3.99 + 18 (24)	1.32	6.99-03 (27)	1.75-21	1.66-10	4.209 + 07
H. B. Robinson	2	V	4.51 + 18	7.43 + 18 (22)	1.65	1.19-02 (25)	1.60-21	1.14-10	1.050 + 08
Surry	1	T	2.50 + 18	2.88 + 18 (9)	1.15	4.56-03 (12)	1.58-21	1.35-10	3.378 + 07
Surry	2	X	3.02 + 18	3.05 + 18 (11)	1.01	4.81-03 (13)	1.58-21	1.30-10	3.687 + 07
North Anna	1	V	2.49 + 18	2.74 + 18 (9)	1.10	4.17-03 (11)	1.52-21	1.17-10	3.570 + 07
Pr. Island	1	V	5.21 + 18	6.09 + 18 (11)	1.17	1.05-02 (16)	1.72-21	2.46-10	4.248 + 07
Pr. Island	2	V	5.49 + 18	6.80 + 18 (10)	1.24	1.19-02 (13)	1.75-21	2.71-10	4.394 + 07
R. E. Ginna	1	R	7.60 + 18	1.17 + 19 (10)	1.54	2.18-02 (14)	1.86-21	2.62-10	8.328 + 07
R. E. Ginna	1	V	4.90 + 18	5.98 + 18 (14)	1.22	1.02-02 (22)	1.71-21	2.22-10	4.612 + 07
Kewaunee		V	5.59 + 18	6.46 + 18 (10)	1.16	1.16-02 (13)	1.80-21	2.86-10	4.057 + 07
Pt. Beach	1	S	--	8.51 + 18 (10)	--	1.48-02 (13)	1.74-21	1.27-10	1.163 + 08
Pt. Beach	1	R	2.22 + 19	2.17 + 19 (10)	0.98	4.41-02 (14)	2.03-21	2.70-10	1.632 + 08
Pt. Beach	2	T	9.45 + 18	9.47 + 18 (10)	1.00	1.59-02 (13)	1.68-21	1.46-10	1.087 + 08
Pt. Beach	2	V	4.74 + 18	7.33 + 18 (11)	1.56	1.23-02 (13)	1.68-21	2.56-10	4.805 + 07
Pt. Beach	2	R	2.01 + 19	2.54 + 19 (10)	1.26	4.68-02 (14)	1.84-21	2.85-10	1.640 + 08
D. C. Cook	1	T	1.80 + 18	2.78 + 18 (22)	1.54	4.61-03 (26)	1.66-21	1.16-10	3.991 + 07
Indian Pt.	2	T	2.02 + 18	3.34 + 18 (22)	1.65	5.49-03 (27)	1.64-21	1.23-10	4.473 + 07
Indian Pt.	3	T	2.92 + 18	3.30 + 18 (22)	1.13	5.38-03 (26)	1.63-21	1.28-10	4.211 + 07
Zion	1	T	1.80 + 18	3.06 + 18 (10)	1.70	4.97-03 (12)	1.62-21	1.31-10	3.789 + 07
Zion	1	U	8.92 + 18	1.02 + 19 (10)	1.14	1.68-02 (13)	1.65-21	1.49-10	1.123 + 08
Zion	2	U	2.00 + 18	2.82 + 18 (9)	1.41	4.54-03 (12)	1.61-21	1.13-10	4.007 + 07
Salem	1	T	2.56 + 18	2.91 + 18 (22)	1.14	4.77-03 (26)	1.64-21	1.39-10	3.426 + 07
Combustion Engineering									
Palisades		A240	4.40 + 19	6.10 + 19 (23)	1.39	9.77-02 (28)	1.60-21	1.37-09	7.130 + 07
Fort Calhoun		W225	5.10 + 18	6.22 + 18 (15)	1.22	9.20-03 (18)	1.48-21	1.12-10	8.191 + 07
Maine Yankee		1	1.30 + 19	1.79 + 19 (19)	1.38	2.43-02 (23)	1.64-21	1.05-09	2.777 + 07
Maine Yankee		2	8.84 + 19	7.85 + 19 (13)	0.89	1.25-01 (18)	1.59-21	8.61-10	1.446 + 08
Maine Yankee		W263	6.90 + 18	6.12 + 18 (13)	0.89	9.21-03 (15)	1.50-21	6.37-11	1.446 + 08
Babcock & Wilcox									
Oconee	1	F	8.70 + 17	7.10 + 17 (21)	0.82	9.83-04 (20)	1.38-21	3.74-11	2.629 + 07
Oconee	1	E	1.50 + 18	1.50 + 18 (10)	1.00	2.11-03 (10)	1.41-21	4.07-11	5.186 + 07
Oconee	2	C	9.43 + 17	1.02 + 18 (10)	1.08	1.50-03 (11)	1.47-21	3.95-11	3.802 + 07
Oconee	3	A	7.39 + 17	8.10 + 17 (10)	1.10	1.15-03 (11)	1.42-21	3.85-11	2.983 + 07
Three Mile Is.	1	E	1.07 + 18	1.09 + 18 (9)	1.02	1.53-03 (9)	1.40-21	3.80-11	4.036 + 07
				avg	1.29				

*Equivalent constant power level exposure time.
**3.17 + 18 reads 3.17 x 10^{18} with a 12% (1σ) uncertainty.

TABLE 3.5

BENCHMARK FACILITIES*, TIME FRAME, PARTICIPANTS, PURPOSE AND USE
(taken from Reference 24)

METALLURGICAL CALCULATIONAL BENCHMARK (IRL-PV)	CALCULATIONAL DOSIMETRY CALIBRATION BENCHMARKS	CALCULATIONAL BENCHMARK (PCA-PV)	METALLURGICAL TESTING BENCHMARK (PSF-PV)	SURVEILLANCE CAPSULE BENCHMARK (PSF-SDMF)	CORE SOURCE BOUNDARY BENCHMARK (VENUS)	PWR CAVITY BENCHMARK (NESDIP)	GENERIC REACTOR BENCHMARKS (BWR-PWR)
1969-1971	1971-2000	1978-1982	1980-1984	1979-2000	1982-1984	1982-1984	1977-2000
NAT. LABS VENDOR	MULTILAB FOR FBR-LWR PROGRAMS	MULTILAB VENDORS, AE, SERVICE LABS	MULTILAB VENDORS, AE, SERVICE LABS	MULTILAB VENDORS, AE, SERVICE LABS	MULTILAB VENDORS, AE, SERVICE LABS	MULTILAB VENDORS, AE, SERVICE LABS	MULTILAB VENDORS, AE, SERVICE LABS
PHYSICS DOSIMETRY METALLURGY SENSOR TESTS	PHYSICS DOSIMETRY SENSOR CALIBRATIONS & QUALITY ASSURANCE	PHYSICS DOSIMETRY SENSOR TESTS & QUALITY ASSURANCE	METALLURGY DOSIMETRY SENSOR LEAD FACTOR TESTS & QUALITY ASSURANCE	SURVEILLANCE CAPSULE PHYSICS DOSIMETRY LEAD FACTOR TESTS & QUALITY ASSURANCE	NEUTRON SOURCE TO SURVEILLANCE & PV WALL POSITIONS LEAD FACTOR IN-VESSEL TESTS	PHYSICS DOSIMETRY SENSOR LEAD FACTOR EX-VESSEL TESTS	PHYSICS DOSIMETRY SENSOR LEAD FACTOR IN-VESSEL EX-VESSEL TESTS

*Acronyms:

AE - Architect-Engineer
IRL-PV - Industrial Research Laboratory Pressure Vessel (PV) Mockup Test.
PCA-PV - Pool Critical Assembly Physics-Dosimetry PV Mockup at ORNL.
PSF-PV - Oak Ridge Research Reactor Pool Side Facility Metallurgical-Dosimetry PV Mockup.
PSF-SDMF - PSF Simulated Dosimetry Measurement Facility.
VENUS - Critical Facility at Mol, Belgium.
NESDIP - NESTOR Reactor Surveillance Dosimetry Improvement Program Ex-Vessel Cavity Mockup at Winfrith, UK.
BWR - Boiling Water Reactor.
PWR - Pressurized Water Reactor.

TABLE 3.6

RELATIVE RATIO FIRST ORR-SDMF RM SENSOR CERTIFICATION TEST*
(X/HEDL)-1 (%) (taken from Reference 42)

Set ID	Reaction	LABORATORY** A	B	C	D	E	F
HNF-1***	^{58}Ni(n,p)	2.38	-7.05	-3.99	2.10	-1.60	-1.77
-3		2.16	-6.34	-3.63	1.37	-2.60	0.24
-2		2.33	-8.59	0.15	-0.93	-2.82	2.69
-4		3.34	-9.24	-0.84	0.47	-2.03	3.90
HNF-1	^{46}Ti(n,p)	2.33	-10.6	3.60	1.43	-0.71	2.16
-3		1.10	-13.9	4.72	2.23	-2.11	1.82
-2		5.72	-7.52	6.98	5.76	-0.85	5.26
-4		4.56	-1.33	7.84	4.56	0.27	3.98
HNF-1	^{63}Cu(n,α)	1.76	-3.38	8.59	-1.12	-2.27	8.05
-3		2.63	1.61	3.05	1.81	-2.00	2.01
-2		1.06	1.40	8.37	3.00	0.59	5.73
-4		4.66	1.85	6.50	2.00	2.14	6.85
HNF-1	^{54}Fe(n,p)	3.02	-6.31	1.95	-3.73	0.39	-5.37
-3		0.56	-10.26	0.11	-2.27	-3.35	-4.13
-2		2.19	-7.63	1.76	1.30	-3.96	0.24
-4		6.49	-7.53	4.69	1.52	0.68	-1.94
HNF-1	^{58}Fe(n,γ)	1.54				1.19	
-3		3.29				0.81	
-2		-4.87				-2.97	
-4		1.96				3.25	
HNF-3	^{59}Co(n,γ)	2.84	-1.55	7.45	1.09	1.83	-1.07
-5		0.06	-7.42	6.36	-1.00	-1.61	-0.52
-4		2.28	-1.84	7.74	1.56	2.82	1.28
-6		1.95	-9.21	6.76	3.72	-0.49	2.35

Set ID	Reaction	LABORATORY** A	B	C	D	E	F
HF-3	^{235}U(n,f)^{140}Ba	0.00			-4.35	-6.68	
HF-5		9.46			-9.40	-4.26	
HF-4		9.39			-7.42	-5.36	
HF-6		-2.92			-14.79	-13.08	
HF-3	^{235}U(n,f)^{103}Ru	5.27		-13.09	-3.18	0.88	
HF-5		8.27		6.21	-9.00	1.54	
HF-4		3.31		-5.59	-2.57	3.17	
HF-6		6.39		-6.37	-8.18	0.99	
HF-3	^{235}U(n,f)^{95}Zr	0.76	0.00	-13.49	4.99	-2.84	
HF-5		5.67	-14.40	9.29	2.26	-3.31	
HF-4		-2.40	-6.49	-11.47	-0.10	-8.36	
HF-6		3.16	-1.78	-6.99	1.54	-5.22	
HF-1	^{237}Np(n,f)^{140}Ba	1.27			-6.38	-11.28	-5.58
HF-2		3.29			-0.91	-13.29	-3.05
HF-1	^{237}Np(n,f)^{103}Ru	3.06		-31.26	-4.37	-0.92	-4.06
HF-2		4.11		-6.94	2.91	-3.28	-2.38
HF-1	^{237}Np(n,f)^{95}Zr	-0.22	10.59	-9.06	4.12	-4.40	-1.46
HF-2		1.99	5.38	-4.76	10.83	-2.80	-1.24
HF-1	^{238}U(n,f)^{140}Ba	2.96			-2.19	-5.60	-0.35
HF-2		0.65			-0.40	-7.05	-1.33
HF-1	^{238}U(n,f)^{103}Ru	5.48		-4.29	-1.93	0.46	-5.17
HF-2		3.74		1.65	2.08	-2.51	-1.79
HF-1	^{238}U(n,f)^{95}Zr	1.72	2.60	-5.81	8.55	-2.56	1.37
HF-2		-1.58	4.41	-3.83	5.35	-6.62	-3.48

*The first RM sensor certification test and the Westinghouse and Combustion Engineering type surveillance capsule perturbation test; fluence (E > 1.0 MeV) of ~6 x 10^{18} n/cm^2 for the thermal shield back (TSB) and ~9 x 10^{17} n/cm^2 for the pressure vessel front (PVF) locations.

**Four vendors and two service laboratories in the U.S. participated in this test. All laboratories remain anonymous for these intercomparisons and are identified only as Laboratories A, B, C, D, E and F.

***HNF-1 and -3 and HF-1, -3 and -5 are the TSB and HNF-2 and -4 and HF-2, -4 and -6 are the PVF locations, respectively.

TABLE 3.7

RELATIVE RATIO SECOND ORR-SDMF RM SENSOR CERTIFICATION TEST*
(X/HEDL)-1 (%) (taken from Reference 42)

Reaction		Laboratory**						
		A	B	C-1	C-2	D	E	F
$^{58}Ni(n,p)$		1.40		- 9.57	- 6.85	-0.96		
$^{63}Cu(n,\alpha)$		0.88		- 3.71	- 2.04	1.84		
$^{54}Fe(n,p)$		1.98		- 7.38	- 3.42	0.75		
$^{58}Fe(n,\gamma)$		0.11		- 2.51	0.22	2.17		
$^{59}Co(n,\gamma)$		1.30		- 4.32	- 1.44	1.65		
$^{237}Np(n,f)$	^{103}Ru	3.42		- 9.70	-10.4			
	^{95}Zr	- 1.58		-10.9	- 5.6			
	^{137}Cs			- 7.83	- 1.34	1.73		
$^{238}U(n,f)$	^{103}Ru	2.09		-11.9	- 8.86			
	^{95}Zr	- 0.78		-11.6	1.58			
	^{137}Cs			-16.8	-7.96	1.38		

*The second RM sensor certification test and the ORR-PSF first simulated surveillance capsule (SSC-1) metallurgical irradiation; fluence (E > 1.0 MeV) of $\sim 2 \times 10^{19}$ n/cm^2.

**Four vendors and two service laboratories in the U.S. participated in this test. All laboratories remain anonymous for these intercomparisons and are identified only as Laboratories A, B, C, D, E and F.

TABLE 3.8

SPECIFIC ACTIVITIES MEASURED BY THE DIFFERENT LABORATORIES
FOR THE ORR-SDMF STARTUP TEST
AND FIRST EUROPEAN LABORATORY RM SENSOR CERTIFICATION TEST
(taken from Reference 43)

	REACTION	SPECIFIC ACTIVITIES RELATIVE TO SCK/CEN				RECOMMENDED SPECIFIC ACTIVITIES ($Bq\ g^{-1}$)	σ (%)
		INTERLABORATORY CAPSULE		AERE/RR & A CAPSULE			
		ECN	PTB	$(AERE)_1$ (1)	$(AERE)_2$ (1)		
SSC	$^{93}Nb(n,n')$	1.17			1.02	$2.062\ 10^7$	9.0
	$^{58}Ni(n,p)$		1.01	1.09	1.05	$7.242\ 10^8$	3.9
	$^{54}Fe(n,p)$	1.01	1.00	1.07	1.10	$1.103\ 10^7$	4.3
	$^{46}Ti(n,p)$	0.99	1.02	1.12	1.07	$8.508\ 10^6$	5.3
	$^{63}Cu(n,\alpha)$	1.02	1.01	0.99(2) (1.29)	(1.05)	$1.201\ 10^5$	1.4
1/4 T	$^{237}Np(n,f)$ ^{95}Zr	0.97	0.98			$3.437\ 10^7$	1.6
	$^{237}Np(n,f)$ ^{137}Cs	0.96	0.98			$2.522\ 10^5$	2.0
	$^{238}U(n,f)$ ^{95}Zr	0.95	0.98			$3.508\ 10^6$	2.6
	$^{238}U(n,f)$ ^{137}Cs	0.99	0.97			$2.738\ 10^4$	1.4
	$^{93}Nb(n,n')$				1.00	$1.330\ 10^6$	0.3
	$^{58}Ni(n,p)$	1.00	1.00	1.07	1.03	$4.472\ 10^7$	3.1
	$^{54}Fe(n,p)$	1.00	0.98	1.11	1.09	$6.956\ 10^5$	6.0
	$^{46}Ti(n,p)$	1.00	1.01	1.12	1.04	$5.851\ 10^5$	4.9
	$^{63}Cu(n,\alpha)$	1.01	1.01	1.01(2) (1.15)	(1.08)	$9.206\ 10^3$	0.5
1/2 T	$^{93}Nb(n,n')$				0.85	$6.643\ 10^5$	11.3
	$^{58}Ni(n,p)$	0.99		1.09	1.02	$1.721\ 10^7$	4.4
	$^{54}Fe(n,p)$	0.97	1.00	1.10	1.10	$2.606\ 10^5$	6.0
	$^{46}Ti(n,p)$	0.98	1.02	1.13	1.07	$2.161\ 10^5$	5.8
	$^{63}Cu(n,\alpha)$	1.03	1.02	1.02(2) (1.37)	(1.30)	$3.465\ 10^3$	1.0
3/4 T	$^{93}Nb(n,n')$				0.84	$3.338\ 10^5$	11.9
	$^{58}Ni(n,p)$	1.00	0.99	1.07	1.00	$6.310\ 10^6$	3.3
	$^{54}Fe(n,p)$	1.00	1.00	1.10	1.07	$9.306\ 10^4$	4.7
	$^{46}Ti(n,p)$	0.96	0.98	(0.76)	1.01	$7.566\ 10^4$	2.2
	$^{63}Cu(n,\alpha)$	1.00	1.01	1.02(2) (1.46)	(1.27)	$1.245\ 10^3$	0.9

(1) $(AERE)_1$: MEASUREMENTS PERFORMED AT HARWELL; $(AERE)_2$: MEASUREMENTS PERFORMED AT WINFRITH

(2) Cu FOIL FROM INTERLABORATORY CAPSULE

TABLE 3.9

POWER REACTORS BEING USED BY LWR-PV-SDIP PARTICIPANTS TO BENCHMARK PHYSICS-DOSIMETRY PROCEDURES AND DATA FOR PRESSURE VESSEL AND SUPPORT STRUCTURE SURVEILLANCE*

(Plant name; reactor type/supplier; reactor operator; ex-vessel cavity (C) and in-vessel (V) surveillance positions available)

Energy Range (MeV)	Type of Dosimeter	Dosimetry Reaction	Nuclear One-1 PWR/B&W Arkansas Power & Light		Nuclear One-2 PWR/CE Arkansas Power & Light		Brown's Ferry-3 BWR/GE Tennessee Valley Authority		H.B. Robinson PWR/WEC Carolina Power & Light		Maine Yankee PWR/CE Maine Yankee Atomic Power		Point Beach-2 PWR/WEC Wisconsin Electric Power		McGuire PWR/WEC Duke Power		CR or DB[i] PWR/B&W		Oconee 1,2&3 PWR/B&W Duke Power[l]		BR-3 PWR/WEC Belgium	
			C	V	C	V	C	V	C	V	C	V	C	V	C	V	C	V	C	V	C	V
>1.5	RMs	^{63}Cu(n,α)^{60}Co[f]	Y		Y	Y	Y	Y	Y	Y	Y	Y		Y	Y	Y	P	Y	P		Y	Y
		^{46}Ti(n,p)^{46}Sc[f]	Y		Y	Y	Y	Y	Y	Y	Y	Y			Y		P	P	PY[j]		Y	Y
		^{54}Fe(n,p)^{54}Mn[f]	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y		Y	Y	Y	P	Y	PY	Y	Y	Y
		^{58}Ni(n,p)^{58}Co[f]	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y		Y	Y	Y	P	Y	PY	Y	Y	Y
		^{238}U(n,f)^{140}Ba-La	Y	(N)	Y	(N)	Y	Y	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)
		^{238}U(n,f)^{103}Ru	(Y)	Y	(Y)	(Y)	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y	Y	Y
		^{238}U(n,f)^{95}Zr-Nb	(Y)	Y	(Y)	(Y)	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y	Y	Y
		^{238}U(n,f)^{137}Cs	(Y)	Y	(Y)	Y	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y	Y	Y
>0.4		^{232}Th(n,f)^{140}Ba-La		(N)		(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	
		^{232}Th(n,f)^{95}Zr-Nb					Y	Y									P	P				
		^{232}Th(n,f)^{137}Cs					Y	Y									P	P				
		^{237}Np(n,f^{140}Ba-La	Y	(N)	Y	(N)	Y	Y	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	P	(N)	(N)	(N)		(N)
		^{237}Np(n,f)^{103}Ru	(Y)	Y	(Y)	(Y)	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y		Y
		^{237}NP(n,f)^{95}Zr-Nb	(Y)	Y	(Y)	(Y)	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y		Y
		^{237}Np(n,f)^{137}Cs	(Y)	Y	(Y)	Y	Y	Y	Y	Y	Y	Y		Y	Y	Y	P	Y	P	Y		Y
		^{93}Nb(n,n')^{93m}Nb								Y	Y	Y					P	P	P			Y
5 x 10^{-7} to 0.5[b]		^{59}Co(n,γ)^{60}Co[e]	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y		Y	Y	Y	P	Y	PY	Y	Y	Y
		^{109}Ag(n,γ)^{110m}Ag[e]	Y		Y	Y	Y	Y	Y	Y	Y	Y			Y		P	P	P		Y	Y
		^{58}Fe(n,γ)^{59}Fe	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y			Y		P	Y	PY	Y	Y	Y
		^{45}Sc(n,γ)^{46}Sc[e]	Y		Y	Y	Y	Y	Y		Y	Y			Y		P	P	PY		Y	Y
		^{235}U(n,f)^{140}Ba-La	Y	(N)	Y	(N)	Y	Y	(N)	(N)	(N)	(N)	(N)	(N)	(N)		(N)	(N)	(N)	(N)		(N)
		^{235}U(n,f)^{103}Ru	(Y)		(Y)		Y	Y	Y	Y	Y	Y			Y		P	P	P			Y
		^{235}U(n,f)^{95}Zr-Nb	(Y)		(Y)		Y	Y	Y	Y	Y	Y			Y		P	P	P			Y
		^{235}U(n,f)^{137}Cs	(Y)		(Y)		Y	Y	Y	Y	Y	Y			Y		P	P	P			Y
As Above[a]	SSTRs[g]	^{238}U(n,f)FP					Y		Y	Y	Y	Y			Y		P	P	P			
		^{232}Th(n,f)FP					Y								Y		P	P	P			
		^{237}Np(n,f)FP					Y		Y	Y	Y	Y			Y		P	P	P			
		^{235}U(n,f)FP					Y		Y	Y	Y	Y			Y		P	P	P		Y	
		^{239}Pu(n,f)FP					Y										P	P	P			
<0.1		^{6}Li(n,α)															P	P	P			
		^{10}B(n,α)															P	P	P			
>0.3		H(n,p) as CR-39		N		N		N		N		N	(N)	N		N	(P)	N	(P)			N
>4	HAFMs[d]	Ni(n,He) as metal[f]							(Y)	Y	(Y)	Y		Y	(Y)	Y	(P)	P	(Y)		(Y)	(Y)
		Al(n,He) as metal[f]	N		N		N		N	Y	(N)	Y		Y	N	Y	N	P	N		N	(Y)
		Cu(n,He) as metal[f]	N		N		N		N	Y	(N)	Y		Y	N	Y	N	P	N		N	(Y)
		Fe(n,He) as metal[f,h]	N		N		N		N	Y	(N)	Y		Y	N	Y	N	P	N		N	(Y)
>0.1[b]		^{7}Li(n,He) as LiF															P	P				
		Be(n,He) as metal								Y	Y						P	P				(Y)
		S(n,He) as PbS									Y						P	P				
		F(n,He) as PbF_2									Y						P	P				
		Ca(n,He) as CaF_2									Y						P	P				
		N(n,He) as NbN or TiN									Y						P	P				(Y)
		Cl(n,He) as $PbCl_2$									Y						P	P				
		O(n,He) as GeO_2															P	P				
<0.1		^{6}Li(n,He) as LiF or alloy								Y	Y						P	P				(Y)
		^{10}B(n,He) nat. or alloy								Y	Y						P	P				(Y)
>0.1	DMs[c]	Quartz															P	P				
		Sapphire								Y							P	P				
		A302B[f,k]								Y				Y			P	P				
		A533B[f,k]								Y				(Y)		Y	P	P				
		Other Steel[f]								Y		Y				Y	P	P				P

*See footnotes for this table on next page.

FOOTNOTES* for Table 3.9:

[a]Energy ranges for the solid state track recorders (SSTRs) are the same as those given for the fissionable radiometric sensors.

[b]Generally these reactions are used with cadmium, cadmium-oxide or gadolinium filters to eliminate their sensitivity to neutrons having energies less than 0.5 eV. The cavity measurements in the Arkansas Power & Light reactors have also included intermediate-energy measurements using thick (1.65 g/cm^2) boron-10 filters (shells) for the ^{235}U, ^{238}U and ^{237}NP fission sensors.

[c]DM means damage monitors (damage to the sensor crystal lattice, such as A302B and A533B or other steels with high copper content and high sensitivity to damage).

[d]HAFM means helium accumulation fluence monitors.

[e]Generally cobalt and silver are included as dilute alloys with aluminum. Scandium is normally ScO_2, and more recently as a ~0.1% ScO_2-MgO ceramic wire.

[f]Frequently when there is no specific HAFM dosimetry package, some of the radiometric sensors and some of the steel damage monitors serve as HAFMs after they have been analyzed for their principal function.

[g]Ni and/or Fe gradient disks were also included in the SSTR capsule, as required.

[h]Iron from RM sensors or Charpy specimens.

[i]Note that power plant CR is Crystal River-3 (Florida Power Corp.) and DB is Davis Besse-1 (Toledo Edison Co.).

[j]The Y following the P refers to a previous Oconee 2 test.

[k]Surveillance capsule reference correlation material (ASTM reference steel plates).

[l]The determination (or feasibility) of using any of the Oconee plants for future benchmark studies has yet to be made.

GE - General Electric
WEC - Westinghouse Electric Company
B&W - Babcock and Wilcox
CE - Combustion Engineering

TABLE 3.10

DOSIMETRY FORMALISM AND METHOD
(taken from Reference 46)

1. DOSIMETRY AND DAMAGE CORRELATION PARAMETERS:

- (EQUIVALENT) FISSION FLUX FOR REACTION $r_i(E)$ IN FIELD $\Phi(E)$:

$$\boxed{\Phi_{f,i} = \int_0^\infty r_i(E)\,\Phi(E)\,dE \Big/ \int_0^\infty r_i(E)\,X_{25}(E)\,dE}$$

WITH $X_{25}(E)$ = URANIUM-235 FISSION SPECTRUM

$r_i(E) = \sigma_i(E)$ FOR DOSIMETERS

$= \varepsilon(E)$ FOR EXPOSURE UNIT ε

- SPECTRAL INDICES

$$\boxed{S_{i,j} = \Phi_{f,i} / \Phi_{f,j}}$$

also $S_{\varepsilon,j} = \Phi_{f,\varepsilon} / \Phi_{f,i}$

2. DOSIMETRY MEASUREMENTS BY BENCHMARK FIELD REFERENCING AT MOL AND NBS CAVITY FISSION SPECTRUM STANDARDS:

$$\boxed{\Phi_{f,i} = \frac{R_i(\text{FIELD})}{R_i(X_{25})} \cdot \Phi_f(X_{25})}$$

R_i = INTEGRAL RESPONSE.

$\Phi_f(X_{25})$ = TOTAL ABSOLUTE STANDARD FLUX DERIVED FROM NBS ^{252}Cf SOURCE STRENGTH

3. DOSIMETRY DATA ANALYSIS BY COMBINATION WITH TRANSPORT THEORY CALCULATIONS:

(EQUIVALENT) " EXPOSURE FISSION FLUX " IS

$$\Phi_{f,\varepsilon} = \Sigma_i\, w_i\, \Phi_{f,\varepsilon}^{(i)}$$

WHERE $\boxed{\Phi_{f,\varepsilon}^{(i)} = (S_{\varepsilon,i})_{CALC.}\, \Phi_{f,i}^{EXPT.}}$

WITH WEIGHT FACTORS

$$W_i = \left| 1-(S_{\varepsilon,i})_{CALC.} \right|^{-1} \sigma^{-2}_{\Phi_{f,i}\,EXPT}$$

AND CORRELATION COEFFICIENTS

$$\rho_{i,j} = \left| 1-(S_{i,j})_{CALC.} / (S_{i,j})_{EXPT.} \right|$$

NOTE THAT $\boxed{\Phi_{f,\varepsilon}^{(j)} / \Phi_{f,\varepsilon}^{(i)} = (S_{i,j})_{CALC.} / (S_{i,j})_{EXPT.}}$

TABLE 3.11

CORRELATION OF FISSION FLUX AND EXPOSURE PARAMETERS
FOR SELECTED PSF NEUTRON DOSIMETERS
(taken from Reference 46)

REACTION COUPLE		$(S_{\varepsilon i} - 1) \ / \ S_{\varepsilon,i}$ CALC.			
ε (EXPOSURE)	i (DOSIMETER)	SSC	1/4T	1/2T	3/4T
DPA STEEL	^{237}Np (n,f)	-9.4%	-7.5%	-6.2%	-3.7%
DPA STEEL	^{103}Rh (n,n')	+5.2%	+9.0%	+12.9%	+16.6%
> 1 MeV	COMPOSITE Rh, In	+4.7%	+1.2%	-3.5%	-9.4%
> 0.1 MeV	DPA GRAPHITE	-1.0%	-0.3%	+0.4%	+0.6%

- DPA STEEL FISSION FLUX AND ^{237}Np FISSION FLUX EQUAL WITHIN $< 10\%$
- $\Phi > 0.1$ MeV AND GRAPHITE FISSION FLUX EQUAL WITHIN $\pm 1\%$

TABLE 3.12

CALCULATED/EXPERIMENTAL SPECTRAL INDEX RATIOS
PSF-HSST MOCKUP (CAPSULE CENTERLINE)
(taken from Reference 46)

	$\frac{^{237}Np(n,f)}{^{27}Al(n,\alpha)}$	$\frac{^{103}Rh(n,n')}{^{27}Al(n,\alpha)}$	$\frac{^{115}In(n,n')}{^{27}Al(n,\alpha)}$	$\frac{^{238}U(n,f)}{^{27}Al(n,\alpha)}$	$\frac{^{58}Ni(n,p)}{^{27}Al(n,\alpha)}$
PSF MOCK-UP					
SSC	-	1.01	1.04	-	1.04
1/4 T	0.89	0.93	0.91	0.89	1.01
1/2 T	0.95	0.89	0.87	0.89	0.97
HSST NORTH					
S-CHANNEL	-	1.11	1.00	-	1.00
C-CHANNEL	-	1.14	1.02	-	1.01
HSST WEST					
S-CHANNEL	-	1.09	0.99	-	1.05
C-CHANNEL	-	1.12	1.01	-	1.02

PSF/HSST EXPOSURE PARAMETERS ($\pm 10\%, 1\sigma$)

	$\phi_{>1MeV}$ [(a)] ($cm^{-2}\ sec^{-1}$)	$\phi_{>0.1MeV} / \phi_{>1MeV}$	$\frac{\text{DPA FISSION FLUX}}{\phi_{>1MeV}}$
ORR-PSF (30 MW)			
SSC	7.70E12	2.77	1.67
1/4 T	5.15E11	3.50	1.90
1/2 T	2.50E11	5.01	2.21
BSR-HSST (2 MW)			
NORTH CAPSULE	9.73E11	3.57	–
WEST CAPSULE	9.21E11	3.59	–

(a) AT MAXIMUM VERTICAL FLUX, NOMINAL CORE POWER AND ON CENTERLINE OF METALLURGICAL EXPOSURE ZONE.

TABLE 3.13

NEUTRON CHARACTERISTICS OF CANDIDATE HAFM SENSOR MATERIALS FOR LWR SURVEILLANCE DOSIMETRY

HAFM Sensor Material	Form*	Principal Helium Production Reaction	Sensor Material Mass* (mg)	Fission Neutron Spectrum: $\bar{\sigma}$ (mb)	Fission Neutron Spectrum: 90% Response (MeV)	Helium** Generation (10^{11} atoms)
Al-0.7%^{6}Li	Alloy Wire	$^6Li(n,\alpha)T$	20	465	0.17 - 5.7	>10,000
Al-0.5% B	(0.05-mm	$^{10}B(n,\alpha)^7Li$	20	499	0.066 - 5.3	>10,000
	diam					
Al	RM foils	$^{27}Al(n,\alpha)^{24}Na$	9	0.69	6.5 - 11.9	1.4
Fe	(3.96-mm	$^{56}Fe(n,\alpha)^{53}Cr$	25	0.33	5.2 - 11.9	0.9
Ni	diam x	$^{58}Ni(n,\alpha)^{55}Fe$	30	4.7	3.9 - 10.1	14
Cu	∿0.25-mm	$^{63}Cu(n,\alpha)^{60}Co$	30	0.54	4.7 - 11.1	1.5
	thick)					
Be	Au/Pt alloy	$^9Be(n,\alpha)^6He$	1	268	2.5 - 7.3	180
NbN	capsules	$^{14}N(n,\alpha)^{11}B$	7	84	1.7 - 5.7	33
GeO_2	(1.27-mm	$^{16}O(n,\alpha)^{13}C$	3	∿12	∿4.0 - 9.0	4.3
PbF_2	diam x	$^{19}F(n,\alpha)^{16}N$	7	24	3.7 - 9.7	8.4
PbS	6.3-mm	$^{32}S(n,\alpha)^{29}Si$	7	∿50	∿0.2 - 5.0	8.8
$PbCl_2$	long)	$^{35}Cl(n,\alpha)^{32}P$	4	13	2.6 - 8.3	2.3
CaF_2		$^{40}Ca(n,\alpha)^{37}Ar$	3	∿30	∿0.1 - 7.4	6.9

*Suggested sensor form and mass for LWR surveillance location. Samples in cavity locations could be made with significantly larger amounts of sensor material, if appropriate.

**Helium generation assuming a fast neutron fluence (> 1 MeV) of 1×10^{18} n/cm^2. For ^{6}Li and ^{10}B isotopes, the helium generation is largely from lower energy neutrons.

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5.0 APPENDIX - RESEARCH INFORMATION LETTER ON AN IMPROVED DAMAGE EXPOSURE UNIT, DPA, FOR LWR PRESSURE VESSEL AND SUPPORT STRUCTURE SURVEILLANCE

Prepared by D. G. Doran, G. L. Guthrie, W. N. McElroy and W. F. Sheeley
HEDL

August 1982

At the request of NRC, HEDL has prepared a Research Information Letter (RIL) on the "DPA Exposure Unit." The RIL points out that the unit for neutron radiation exposure commonly used in assessing reactor pressure vessel embrittlement and fracture toughness (neutron fluence above 1 MeV) is not conservative when applied to the prediction of steel property degradation in the outer thicknesses of the pressure vessel and in its support structures. Further, it concludes that the use of dpa is now well established. DPA is being recommended in the appropriate new ASTM LWR standards as an improved neutron exposure parameter for use by utilities, vendors, service and research laboratories, and licensing and regulatory agencies.

The RIL is reproduced in this appendix for reference purposes and comments by LWR-PV-SDIP participants are solicited; particularly to new reference information related to the use and application of the dpa exposure parameter.

Introduction

The purpose of this Research Information Letter is: 1) to point out that the measure of neutron radiation exposure (neutron fluence above 1 MeV, hereafter called "fast fluence") commonly used in assessing reactor pressure vessel embrittlement is not always conservative, 2) to propose a better measure of damage exposure, displacements per atom (dpa), and 3) to give evidence to support the adoption of dpa. The replacement of fast fluence by dpa has particular significance in the analysis of thermal shock.

Background

The prediction of property changes in a reactor pressure vessel during service is of major importance in assuring reactor safety. This prediction is normally accomplished by measuring the property changes of the same steel subjected to accelerated exposure tests and developing a property change--exposure correlation. Then, using neutron physics and dosimetry methods to determine the neutron exposure at the location of interest, the correlation is used to predict the property change at that location.

The incremental property change, specifically, the increase in nil ductility transition temperature (NDTT), and the decrease in fracture toughness that results from a single neutron penetrating a steel pressure vessel depends on the energy of the neutron. Mechanical property degradation results when the neutron dissipates its energy by displacing atoms from their normal sites. This allows the formation of small defect clusters that inhibit plastic flow and create a brittle condition in the alloy. The number of atoms displaced by a neutron, and therefore the amount of property degradation, generally increases with the energy of the neutron. The shape of the neutron energy spectrum (neutrons per unit energy plotted vs energy) varies among reactor types and from one position to another within a given reactor (see Table 1). Therefore, when correlating property degradation with neutron fluence, the energy spectrum to which the material is exposed must be properly taken into account.

The current regulatory practice in expressing material exposure is to count only those neutrons having an energy greater than 1 MeV [n/cm^2 (E > 1 MeV)]. Although use of fast fluence is an empirical correlation procedure, its effect is to weigh all neutrons above 1 MeV equally and give zero weight to all lower energy neutrons.

The dpa unit of exposure was developed to more properly account for the neutron energy spectrum. A dpa value is readily assigned to each exposure, when the neutron energy spectrum is known, by weighting each neutron by its ability to produce atomic displacements.[1] The calculation of the weighting function (the displacement cross section) requires only a knowledge of nuclear reactions and energy loss mechanisms and the application of widely-accepted procedures. No property degradation information is used. The dpa is thus a spectrum-sensitive exposure unit, which is independent of flux level, temperature, minor alloying elements, and material property.

The dpa, because of its physical basis, should correlate property data obtained in different spectra better than the fluence greater than 1 MeV. The dpa exposure unit will serve as the basis for improved damage correlation parameters, which may be developed as additional data are obtained. For example, thermal neutrons will react strongly with boron present in the steel. Any property degradation that may result is not represented by the displacement cross section. Similarly, the effects of damage rate, if any, are not accounted for. This may be important in some pressure vessel steels--higher flux levels and shorter exposure times may produce more damage per dpa than lower flux levels and longer exposure times. If this turns out to be an important consideration, it will be taken into account through damage models that incorporate both the spectral effect, through use of dpa or a modification thereof, and a rate effect.

Supporting Evidence

Evidence has been developed in a variety of programs for the superiority of dpa over fast fluence as a damage exposure unit. A case can be made based solely on LWR data on embrittlement, but the data have been obtained over a relatively narrow range of neutron spectra. Therefore, some analyses of other water reactor and breeder reactor data and data obtained with fusion energy (14 MeV) neutrons are also included. While these data are generally for tensile properties, rather than embrittlement, and for temperature regimes both higher and lower than LWR operating temperatures, their inclusion in the present context is appropriate.

Several analyses of the spectral dependence of embrittlement in pressure vessel steels have been made recently. Simons[2] reevaluated data developed by Serpan and McElroy[3-7] on the shift in ductile-brittle transition temperature (DBTT) for five data sets on three steels using a simple correlation model suggested by Odette.[8] Statistical analyses of fits to the data using dpa, fast fluence, and three other exposure indices were performed. The relative variances for dpa and fast fluence are shown in Table 2. Clearly, dpa gives a better overall correlation of the data than does fast fluence.

Odette[8] fit available data on ΔDBTT for several materials for irradiation temperatures below 240°C (464°F) to a simple correlation model using either dpa or fast fluence as the exposure unit. Comparative plots of measurement vs predicted values (from his correlation model) are shown in Figures 1a and b. The superiority of dpa as a correlation parameter is again obvious.

Mas et al.[9] measured the ΔDBTT in A508 steel irradiated at 235°C (455°F) in different neutron spectra to approximately the same fast fluence. The degraded and softer steel block PV simulator spectrum (PCBT), which had an enhanced low energy (0.1 to 2 MeV) neutron component, produced 30% to 40% greater ΔDBTT than the less degraded and harder core edge in steel spectra (61 AVD and 73 AVG), see Table 1, French Melusine test facility. Their "probable zones" model, which is very similar to the displacement cross section, and fluence (E > 0.1 MeV) correlated the data significantly better than fast fluence. As is often the case, the duration of the "softer spectrum" irradiation was much longer (about a factor of 10) than the "harder spectra" irradiations, raising the possibility of a damage rate effect.

Alberman et al.[10] measured the increase in yield strength in nonboron-containing A533B steel irradiated at 100°C (212°F) in two very different neutron spectra, in test locations in a light water and in a heavy water research reactor. The softer heavy water spectrum had an enhanced (0.1 to 2 MeV) neutron component. For equal increases in yield strength, the harder spectrum light water irradiation required about 60% more fast neutron exposure. Of several damage models studied, the "probable zones" and "EURATOM dpa" iron cross sections provided good correlation of the experimental results. (For such fission reactor spectra, the EURATOM and ASTM iron displacement cross sections give calculated dpa results that are in good agreement within a few percent. Consequently, either cross section can be used with equal confidence.)

A spectral effects experiment was carried out in the breeder reactor program. Blackburn et al.[11] measured tensile properties of several austenitic steels irradiated at 385°C (725°F) in a range of neutron spectra in the fast reactor EBR-II. Of several exposure indices examined (not including fluence above 1 MeV), dpa best characterized the spectral dependence of the data. Fast fluence would have given a poor correlation, overpredicting the observed spectral effects by about a factor of two.

An earlier study of available data on the yield strength of 304 stainless steel by Simons et al.[12] showed a 40% reduction in standard deviation when fast fluence was replaced by dpa. The correlation of other tensile properties was also improved.

The fusion program has prompted several experiments to compare changes in tensile properties in specimens irradiated with either fission neutrons or fusion neutrons, generally near room temperature. While the results vary with material and property, the correlation is always improved by replacing fluence greater than 1 MeV by dpa.[13]

Evaluation and Application

The importance of the thermal shock problem has focused attention on the prediction of damage exposures through the pressure vessel wall. It is essential that such predictions be conservative. Recent calculations[8,14] have shown that the ratio of dpa to fast fluence increases by about a factor of two in a traverse from the inside to the outside of the vessel wall. Some of these results are given in Table 3.[14] [They are consistent with earlier calculations for the ORR-PSF (PV Mockup) included in Table 1.] This is a clear case where the spectral effect is important. In particular, if fracture toughness does indeed correlate better with dpa than with fast fluence, as the evidence strongly suggests, predictions of the fracture toughness deep in the vessel wall will not be conservative if they are based on fast fluence.

The dpa unit is widely used in the breeder and fusion materials programs. IAEA Specialist's Meetings, EURATOM, and ASTM have adopted dpa as the accepted international unit for reporting neutron exposures and have recommended standard methods of calculating it (see Attachment 1). A recent IAEA Advisory Group Meeting workshop specifically recommended the use of dpa for use in damage evaluation of pressure vessel steels (see Attachment 2).

A set of ASTM standards is currently being developed to cover improved neutron physics-dosimetry-metallurgy surveillance of LWR pressure vessels and support structures (see Figures 2 and 3). The dpa is adopted as the standard exposure unit in these standards.

An ASTM "Recommended Pratice for Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements per Atom (dpa)" has been written (ASTM Standard E693-79).[1] It contains a table giving the displacement production cross section in a ferritic steel for neutrons in the energy range from 1×10^{-4} eV to 10 MeV. The complex structure (see plot in Figure 4) simply reflects resonances in the nuclear reaction cross sections and is independent of any assumptions regarding displacement mechanisms. At energies less than 5×10^{-4} MeV, the calculated displacements are all due to the (n,γ) reaction. At LWR operating temperatures, it is unlikely that this reaction contributes significantly to property degradation. However, the early work of Serpan et al.[4-7] on A302B steel irradiated below 240°F (116°C) indicated a thermal neutron effect for reactor locations when the thermal-to-fast ratio was greater than about 10. More recent experiments by Alberman et al.[10] showed a thermal effect in A533B containing up to 5-ppm boron irradiated at 100°C (212°F). Boron-free specimens showed no effects. Thus, the role of low energy neutrons in mechanical property degradation is not yet clear. This is of no concern for pressure vessel spectra because there are so few low energy neutrons. It may possibly be significant in analyses of some support structure damage exposures and in some test reactor spectra with very high thermal-to-fast ratios.

The displacement cross section tabulated in ASTM Standard E693-79[1] can be used together with the neutron spectrum corresponding to a specific reactor location to calculate the rate of accumulation of displacements. Two other new related standards have been written and will be available from ASTM in early 1983. One is ASTM Standard E706 (IID), "Application of Neutron Transport Methods for Reactor Vessel Surveillance," which describes the recommended procedures and data to be used for obtaining the required calculated multigroup fluxes. The other is ASTM Standard E706 (IIA), "Application of Neutron Spectrum Adjustment Methods," which describes recommended procedures for obtaining dosimetry-adjusted values of these calculated multigroup fluxes. Supporting these two ASTM reactor physics calculational guides are an ASTM Nuclear Data Guide E706 (IIB), "Application of ENDF/A Cross Section and Uncertainty Files," and an ASTM Dosimetry Guide E706 (IIC), "Sensor Set Design and Irradiation for Reactor Surveillance," and a set of five "Sensor Measurement Method" standards, ASTM E706 (IIIA), III(B), III(C), III(D), and III(E). The IIB, IIC, IIIA, IIIB, and IIIC standards are now or will be available from ASTM in early 1983 and will be included in the 1983 Annual Book of ASTM Standards. The III(D) and III(E) standards will be available at a later time, see Figures 2 and 3.

The adoption of dpa as the exposure index for use in damage correlations deals only with the effect of neutron spectrum on damage. Efforts must continue to properly account for differences in material chemistry, damage rates, irradiation temperatures, and exposure times, as clearly pointed out by Randall.[15] A new ASTM Standard E706 I(E), "Damage Correlation for Reactor Vessel Surveillance," is being prepared that will address these variables, see Figures 2 and 3.

Recommendations

It is strongly recommended that dpa be adopted as the exposure index for radiation-induced changes in the properties of pressure vessel and support structure steels. (Values of fluence above 1 MeV should also be reported.) This means that it is imperative that neutron flux-fluence-spectra be determined accurately in locations where materials data are obtained and where they are to be applied.

It is further recommended that NRC accept the applicable ASTM Standards, E706 (ID) [E693-79], E706 (IIA), E706 (IIB), E706 (IIC), E706 (IID), E706 (IIIA), E706 (IIIB), and E706 (IIIC), as the basis for the employment of dpa as an exposure index.

TABLE 1*

METALLURGICAL STEEL IRRADIATIONS--TEST FACILITIES ESTIMATED ENVIRONMENTAL PARAMETER VALUES FOR ASSESSING TEMPERATURE, FLUX LEVEL, AND SPECTRAL EFFECTS

E 708

TABLE 3 Metallurgical Steel Irradiations—Test Facilities Estimated Environmental Parameter Values for Assessing Temperature, Flux Level, and Spectral Effects

Test Facility	Irradiation Temperature	$\left[\frac{\bar{\sigma}\ \text{Fe DPA}}{\phi(>1.0\ \text{MeV})}\right]$	$\left[\frac{\bar{\sigma}\ Fe^{54}(n, p)}{\phi(>1.0\ \text{MeV})}\right]$	$\left[\frac{\phi(>0.1\ \text{MeV})}{\phi(>1.0\ \text{MeV})}\right]$	Flux (>1.0 MeV) Level
	°C	(barns)	(Milli-barns)		(n/cm²·sec)
United States:					(peak)
ORR-PSF (PV Mockup)[A] (30 MW)					
Core center-in fuel		1390[B]	107[C]	1.85	$3.8 + 10^{14}$
Surveillance capsule-center	288 ± 5	1606	66	3.08	6.8×10^{13}
PV surface position-incident	"	1588	83	2.79	9.6×10^{11}
PV T/4 position	"	1844	61	4.10	4.6×10^{11}
PV T/2 position	"	2238	49	5.77	2.1×10^{11}
PV 3T/4 position	"	2727	41	7.82	9.8×10^{10}
PV void box back face position	ambient (~44°C)	3060	41	8.82	3.1×10^{10}
ORR-30, BSR-2MW					
Core center-in fuel	288 ± 5	1386[B]	107[C]	1.84	3.2×10^{13}
Surface position center 4T HSST specimen	288 ± 10	1623	64	3.01	2.1×10^{13}
Buffalo (2 MW)					
Core positions in NRL/NRC steel test assembly	288 ± 5	1765[B]	73[C]	3.54	$\sim 3 \times 10^{13}$
Virginia (2 MW)					
Core position in Westinghouse/EPRI steel test assembly	288 ± 5	~1765	~73	~3.50	$\sim 3 \times 10^{13}$
PWR					
Surveillance position-in steel capsule	288–321	1800[B, D]	63[C, D]	...	$0.1–2.0 \times 10^{11}$
PV surface	288–321	1630	110	...	$0.07–0.70 \times 10^{11}$
PV T/4 position	288–321	1960	78	...	$0.04–0.40 \times 10^{11}$
Ex-vessel cavity	ambient	...	...	...	$0.0020–0.20 \times 10^{11}$
BWR					
Surveillance position					
PV surface	280	1553[B]	182[E]	~1.6	$10^{8}–10^{9}$
PV T/4 position	270	1603	141	~2.5	$\sim 10^{8}$
Ex-vessel cavity	55	2014	98	~4.4	$10^{7}–10^{8}$
United Kingdom:					
DIDO (25 MW)					
Core center (in fuel)	XXX ± 5	1600[B] (1567)[F]	112[C]	2.3	5×10^{13}
2V4 position (steel)	XXX ± 5	1962 (1665)	133	2.3	3×10^{13}
Pluto (25 MW)					
Core center (in fuel)	XXX ± 5	1646 (1594)	117	2.3	5×10^{13}
CA6B position (steel)	XXX ± 5	3094 (1700)	168	2.0	5×10^{11}
Herald (5 MW)					
Core center (in fuel)	XXX ±	1367 (1356)	102	1.9	3.5×10^{12}
DIO position (steel)	XXX ± 5	1457 (1451)	86	2.8	3.5×10^{12}

*1982 Annual Book of ASTM Standards, Volume 45.

TABLE 3 *Continued*

Test Facility	Irradiation Temperature	$\left[\frac{\bar{\sigma}\ \text{Fe DPA}}{\phi(>1.0\ \text{MeV})}\right]$	$\left[\frac{\bar{\sigma}\ \text{Fe}^{54}(n,p)}{\phi(>1.0\ \text{MeV})}\right]$	$\left[\frac{\phi(>0.1\ \text{MeV})}{\phi(>1.0\ \text{MeV})}\right]$	Flux (>1.0 MeV) Level
	°C	(barns)	(Milli-barns)		($n/cm^2 \cdot sec$)
France:					
Melusine					
PV-simulator (PCBT position)	235 ± 5	1970	...	4.31	1.8×10^{12}
Core edge in steel (61 AVD)	235 ± 5	1450	...	2.13	1.9×10^{13}
Core edge in steel (73 AVG)	235 ± 5	1450	...	2.14	1.9×10^{13}
Germany:					
FRJ 1					
Core edge (G7)-without steel	290 ± 5		122^{B}	1.85^{B}	$1.51 \times 10^{13\,B}$
FRJ2					
Core (B5)	290 ± 5	1480^{H}	88	2.29	4.8×10^{13}
Reflector 2V4-with steel	290 ± 5	1640	83	3.06	3.7×10^{13}
Belgium:					
BR2					
DGR position (core)	~150	1390^{G}	87^{C}	2.1	$\sim 3 \times 10^{14}$
BR3					
Surveillance position	260^{I}	1422^{G}	120^{C}	1.9	$8.8 \times 10^{12\,J}$
Dosimetry position in reflector	260^{I}	1410	126	1.8	2.3×10^{12}
PVF (in water)	260^{I}	1621	94	2.9	$1.7 \times 10^{11\,J}$
PVF (in steel)	260	1645	88	3.1	$1.5 \times 10^{11\,J}$
PV - T/4	260	1723	75	3.6	$1.2 \times 10^{11\,J}$
Between PV and Neutron Shield Tank	60	2314	50	6.1	$2.2 \times 10^{10\,J}$
Japan:					
JMTR (50MW)					
Core center	270–320	...	81^{C}	1.87	$1.0\text{–}2.0 \times 10^{14}$
Core position in steel	"	1320^{C}	71	1.9	1.0×10^{13}
Reflector-IAEA Steel position	"	...	65	2.5	1.0×10^{13}
JRR-2 (10 MW)					
Core (6D)-in Hollow Fuel	Ambient (~50°C)	1590	104^{L} (100)	2.4	2.3×10^{13}
Core center (VT-1) Position in flux trap	"	1780 $(1590)^{K}$	113^{K} (109)	3.0	1.8×10^{13}
Netherlands:					
HFR (45 MW) E5 core position	250–650	1316^{L}	90	1.87	2.1×10^{14}

[A] 4/12 Configuration (4 cm H_2O between core and thermal shield and 12 cm of H_2O between the thermal shield and PV wall).
[B] ASTM E 693 – 79 iron $\sigma(E)$ with (n, γ) recoil displacements included.
[C] ENDF/B-IV $\sigma(E)$.
[D] Perturbed spectrum (40°) for a 4-loop reactor geometry.
[E] ENDF/B-V $\sigma(E)$.
[F] Without (n, γ) recoil displacement included.
[G] Doran and Graves iron displacement cross section, HEDL SA-1058 (1976).
[H] Spectrum unfolding by L. Wiese, with more appropriate two-dimensional input spectrum.
[I] Temperature of water.
[J] Based on calculation BR 3/core 4A.
[K] JENDL-1 $\sigma(E)$.
[L] Damage cross section library DANSIG-77, E CN-36 report.

TABLE 2

A COMPARISON OF VARIANCES FOR FITS
OF ΔDBTT TO DIFFERENT EXPOSURE INDICES[2]

Steel	Irradiation Temp (°F)	No. of Spectra	Relative Variance dpa	Relative Variance $\phi t > 1$ MeV
A212B	284	3	1.00	1.55
A302B (LT)	$\leq$450	5	1.00	2.26
A302B (HT)	550 to 585	5	1.00	1.24
A350 LF1	430	2	1.14	1.50
A350 LF3	510	2	2.89	3.59

TABLE 3

dpa/FLUENCE (E > 1.0 MeV) IN UNITS OF 10^{-21} dpa/(n/cm^2)

Reactor	Supplier	Accelerated Surveillance Capsule	Accelerated Surveillance Capsule Positon (deg)	Wall Capsule	Wall Capsule Position (deg)	Wall Traverse 0 T	1/4 T	1/2 T	3/4 T	1 T	Azimuthal Angle of Wall Traverse (deg)
2-Loop	W	1.80	13	None		1.59	1.79	2.06	2.36	2.56	13
		1.73	33								
3-Loop Shield	W	1.58	15	None		1.57	1.84	2.25	2.72	3.05	15
4-Loop Shield	W	1.58	40	None		1.61	1.89	2.37	2.95	3.36	40
3-Loop Pad	W	1.95	19.7	None		1.61	1.90	2.37	2.90	3.27	19.9
4-Loop Pad	W	2.00	31.5	None		1.58	1.90	2.39	3.02	3.46	31.5
	CE	1.49	35	1.47	3	1.51	1.78	2.23	2.82	3.31	35
Type 177 Fuel Assembly	B&W	1.47	11	1.37	11	1.51	1.70	NC	NC	3.00	9.8

B&W - Babcock & Wilcox.
CE - Combustion Engineering.
NC - Not Calculated.

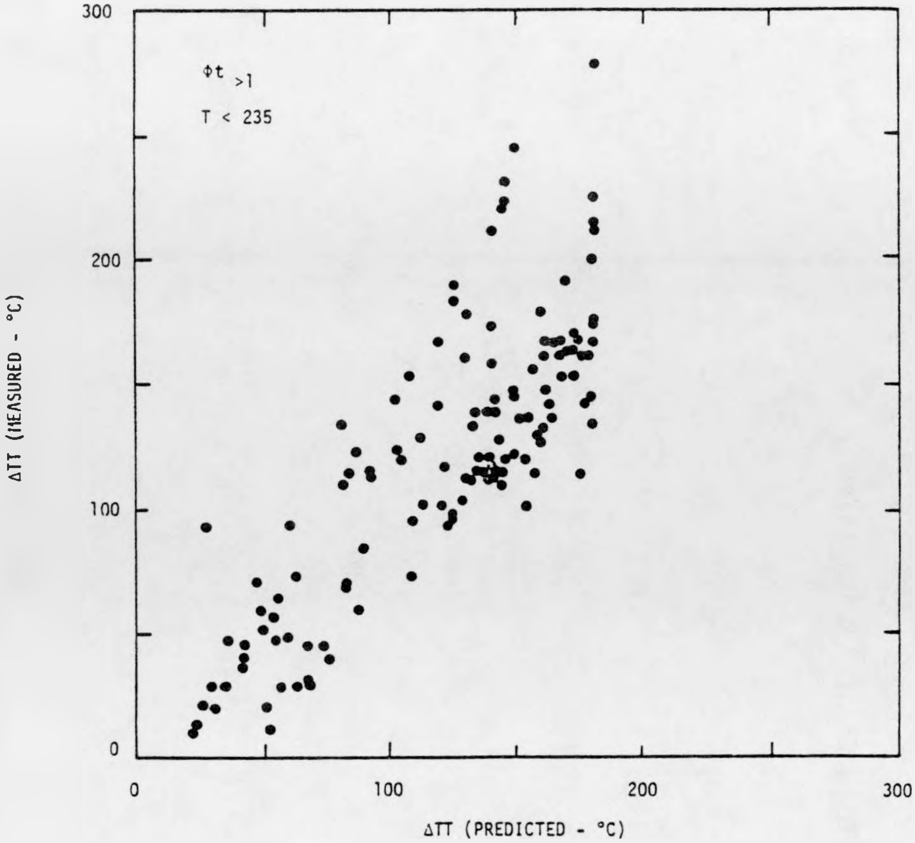


FIGURE 1a. Correlation Based on Fluence >1 MeV.(8)

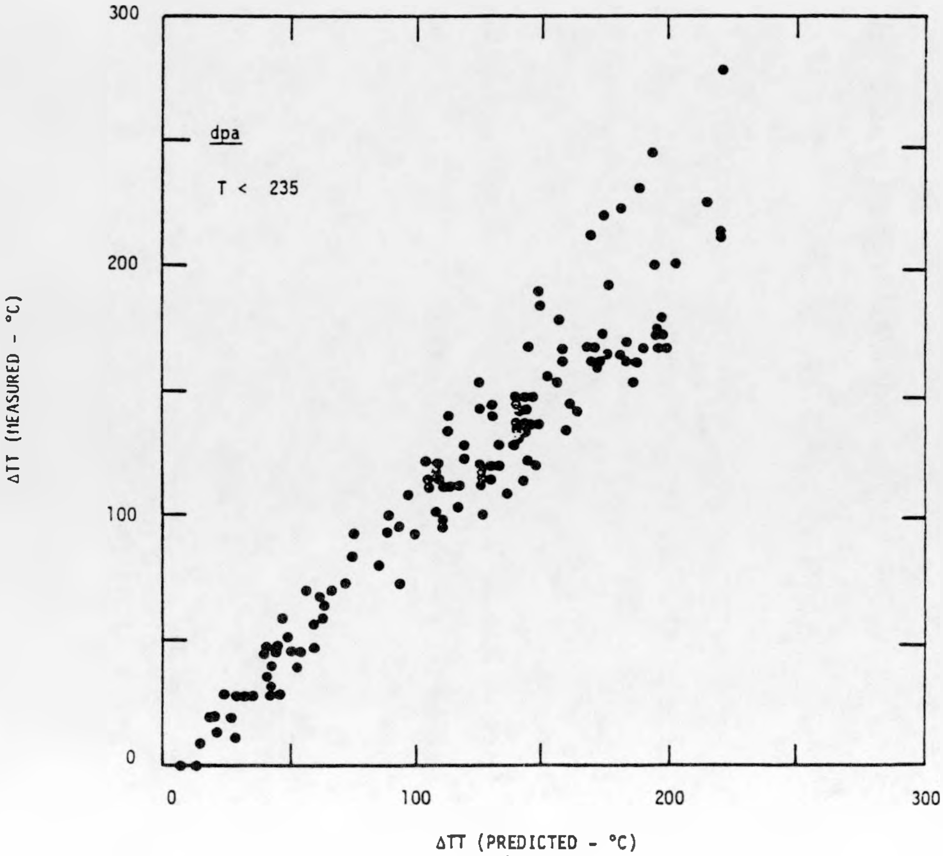
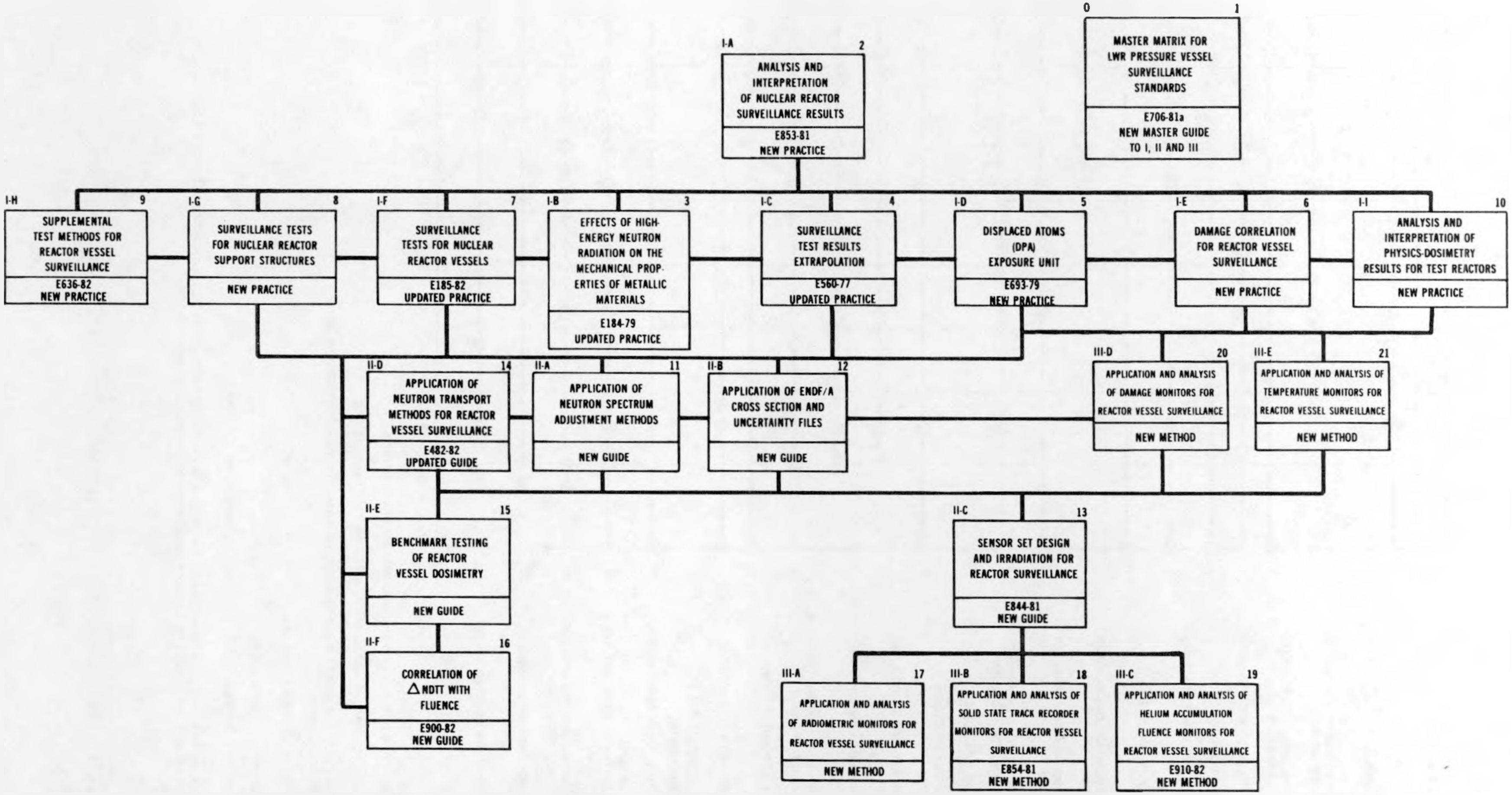


FIGURE 1b. Correlation Based on dpa.(8)



ASTM STANDARDS FOR SURVEILLANCE OF LWR NUCLEAR REACTOR PRESSURE VESSELS AND SUPPORT STRUCTURES.

FIGURE 2. ASTM Standards for Surveillance of LWR Nuclear Reactor Pressure Vessels and Support Structures.

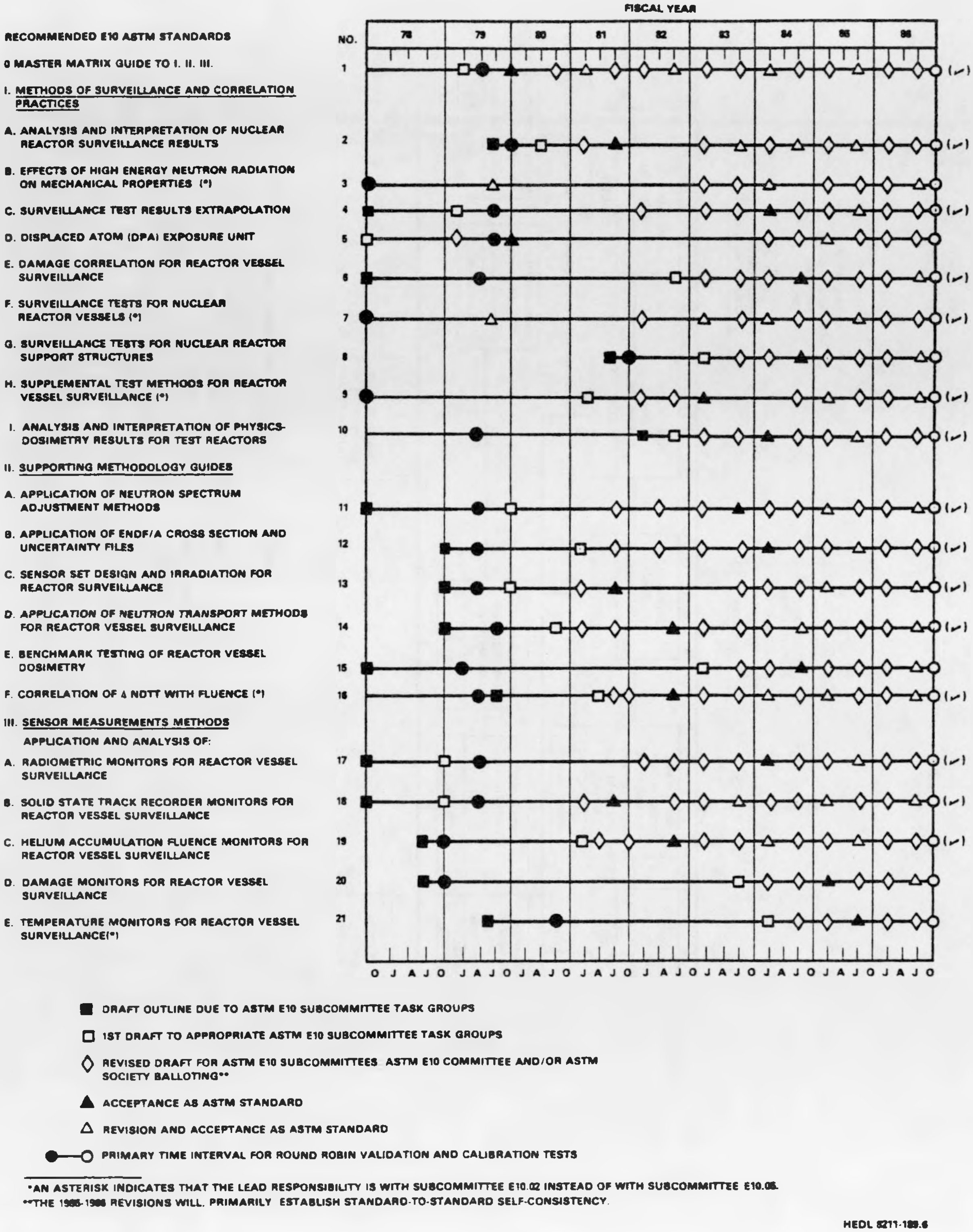


FIGURE 3. ASTM LWR Standards Preparation Schedule.

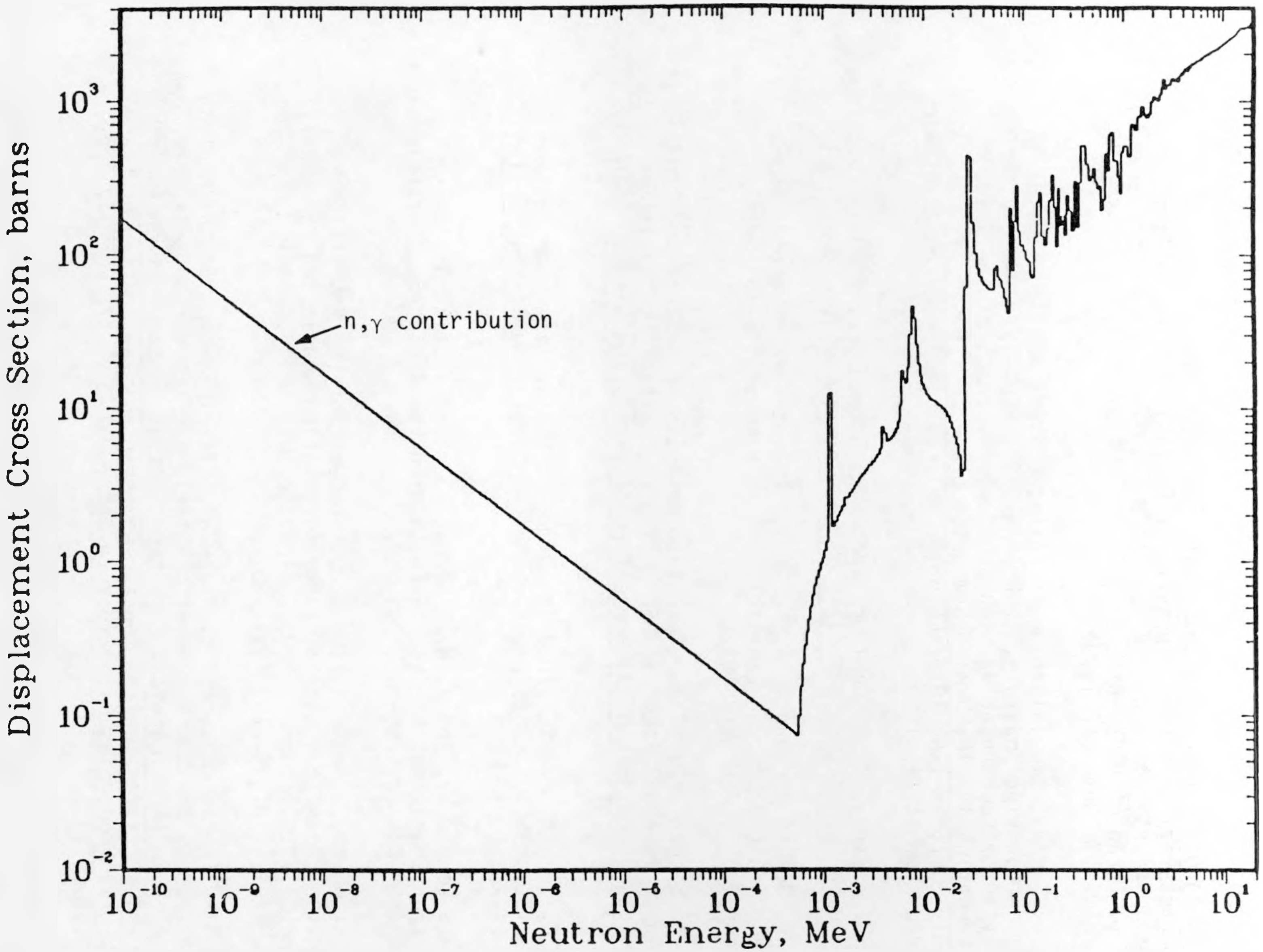


FIGURE 4. Displacement Cross Section for Iron, Plotted as a Function of Neutron Energy.

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ATTACHMENT 1 *

LETTERS TO THE EDITOR

RECOMMENDATIONS FOR THE CALCULATION OF MATERIALS IRRADIATION EXPOSURE

At a specialists' meeting on radiation damage units, held at Harwell, United Kingdom, November 2-4, 1976, within the program of the International Atomic Energy Agency (IAEA) International Working Group on Reactor Radiation Measurements (IWGRRM), recommendations were endorsed by the international group of experts attending the meeting. Publication of these recommendations in *Nuclear Technology* will assist in their dissemination throughout the nuclear community and in achieving standardization of atomic displacement calculations by groups in different countries.

RECOMMENDATIONS FOR THE CALCULATION OF MATERIALS IRRADIATION EXPOSURE

Continued use of the 1972 recommendations on atomic displacement calculations in metals[1] is recommended with the following clarifications. Comparisons have shown that two sets of damage energy cross sections for iron, chromium, and nickel (and hence steels), calculated according to these recommendations and based, respectively, on UKNDF (Ref. 2) and ENDF/B-IV (Ref. 3) reaction cross-section files, agree to within adequate accuracy[a] when applied to fission reactor spectra. Since there is no sound basis or practical significance for selecting one set over the other, it is recommended that one or the other be used for displacement per atom (dpa) calculations. It is recommended that the Neutron Data Centers at Brookhaven, Obninsk, Saclay, and Vienna be asked to maintain and make available the above damage cross sections in the 31-group MUFT structure and, in the case of the Ref. 2 data, in the 621-group SAND-2 structure as well.

Comparisons of damage energy cross sections for zirconium tabulated in Refs. 1 and 2 exhibit unsatisfactorily large discrepancies that must be resolved. In the interim, it is recommended that both damage energy cross sections also be made available. It is further recommended that the conversion from damage energy to displacements for zirconium follow the same prescription as for iron.

No damage energy cross sections are recommended at this time for application to neutron spectra harder than a fission spectrum, such as are of interest in fusion reactor development programs. Further comparisons of cross-section sets extending to high energies must be made. It is recommended that within the next two or three years, exchanges of reevaluated damage energy cross sections be made with the objective of resolving remaining differences.

In conclusion, we recommend the continued use of dpa as a spectrum sensitive measure of a material's irradiation exposure but would emphasize that dpa should not be interpreted as a direct measure of actual defect damage in the material.

[a]The agreement for nickel and iron is within a few percent; the agreement for chromium is somewhat poorer, but the discrepancy is negligible in applications to stainless steels.

RECOMMENDATIONS FOR THE ATOMIC DISPLACEMENT CALCULATIONS IN METALS

These recommendations apply to atomic displacement calculations in metals. A principal objective is the formation of a basis for the uniform reporting of neutron and other particle irradiation damage exposures in the study of irradiation effects in metals. We make the following recommendations:

1. For all irradiations, the experimental conditions and how they were determined should be fully specified. This includes:

   a. the reactor and location within the reactor, the neutron flux, the neutron spectrum, and the irradiation time and temperature, or

   b. the ion species, the ion energy and particle flux, the irradiation time and temperature, the method of irradiation (scanning, rocking, etc.), the depth at which the sample is taken, and the sampling thickness and crystal orientation where relevant, or

   c. the electron energy, the displacement cross section, the electron flux, the irradiation time and temperature, the foil thickness, and crystal orientation.

2. In addition to the above data, we recommend that the irradiation exposure be quoted in terms of dpa, using the following interim procedure for calculating secondary displacements[4]:

$$N_d = \beta E_{\mathrm{Damage}} \ , \qquad \text{displacements/primary} \ ,$$

where $\beta = 10\ \mathrm{keV}^{-1}$ for iron, steels, and nickel-based alloys and E_{Damage} is an estimation of energy deposited into atomic processes given by

$$E_{\mathrm{Damage}} = \frac{E}{[1 + kg(\epsilon)]}$$

$$k = 0.1337\ Z^{2/3}/A^{1/2}$$

$$\epsilon = E/[86.931\ Z^{7/3}] \qquad (E \text{ in eV}) \ ,$$

where Z and A are the atomic and mass numbers, respectively.

For neutron irradiation, the relevant neutron cross section, the reaction kinematics, and spectral data used for calculating the primary recoil spectrum should be referenced. In the case of ion bombardment, the method of calculating the energy deposited into atomic processes (E_{Damage}) as a function of depth should be stated with appropriate definition of parameters.

3. Future work should include studies of the energy partition and recombination processes. Recognizing the dependence of displacement calculations on neutron interaction cross sections, we recommend that the IAEA compile and evaluate cross-section sets used in such calculations.

RECOMMENDATIONS FOR GRAPHITE

The meeting saw no reason to change the conclusions reached at the Seattle meeting in 1972, and it agreed that the recommendations made at that meeting and also published in Ref. 2 should continue to be used.

Thank you for your cooperation in publishing these recommendations.

V. Chernyshev, Scientific Secretary

International Atomic Energy Agency
International Working Group on
Reactor Radiation Measurements
A-1011 Vienna, Austria

November 11, 1977

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ATTACHMENT 2

In the Proceedings of an Advisory Group Meeting on Nuclear Data for Radiation Damage Assessment and Related Safety Aspects, IAEA-TECDOC-263, 1982, p. 330, the Chairman, W. Schneider, of "Workshop 2: Status of Nuclear Data for Radiation Damage Calculations and Damage Correlation Estimates," indicated that the participants discussed the quality and availability of displacement cross-sectional data. The following conclusions and recommendations were made:

1) It is recommended, for the time being, to use the following sets of displacement cross sections: ASTM and EURATOM (for 640 neutron energy groups and for neutron energies up to 20 MeV). The data sets are based on ENDF/B-IV (ASTM) and ENDF/B-III (EURATOM) libraries. These sets have been published in: ASTM Standard E693-79 and included in EUR 5274 (in 50 energy groups); the EURATOM set will be published in the DAMSIG-81 data library (in 640 groups)*. These recommended sets should be applied particularly for damage evaluation for pressure vessel steels in light water reactors.

2) It is recommended to develop a new Reactor Radiation Damage Nuclear Data File of an international reference status within the next three years. This file should incorporate the file being prepared now in the USA which is based on ENDF data and which is expected to be issued in 1982 and made available internationally through the four nuclear data centers. It is understood that the released US-file will include data for Fe, Cr, and Ni up to 20 MeV.

   It is recommended to supplement the future International Reactor Radiation Damage File for Fe, Cr, and Ni up to 40 MeV and to include the data for Al up to 40 MeV with the first priority. The data for Graphite, O, Ti, V, Mn, Cu, Zr, Mo, W up to 40 MeV and for Nb, Sn up to 20 MeV should be included in the file with second priority.

   Few experimental data above 20 MeV exist. More experimental data are wanted, but in their absence one has to recur to theoretical calculations. Theoretical calculations of H and He production cross sections show that at higher incident energies the contributions of reactions of the type (n,pp) and $(n,p\alpha)$ cannot be neglected for target nuclei with small neutron excess. Evaluations of needed changes in the energy dependence of the damage function should be considered in future theoretical and experimental research. For special purposes and environments (e.g., D_2O reactors, strong γ-ray fields) the file should include the damage cross sections for (n,γ), (γ,n) and (γ,γ') reactions.

*Available as ECN-104, Petten, November 1981.

3) The Working Group recognizes the importance of neutron and gamma-ray kerma as a damage mechanism in organic materials such as fiberglass-reinforced epoxy. Some nuclear data for separate isotopes are required along with Q-values, nuclear decay data, and spectra of emitted particles. The data for the following elements are essential in this context: Be, C, N, O, F, Na, Al, Si, Ca. The Working Group strongly supports the recommendations of the IAEA Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology, December 1978 [INDC(NDS)-101/LF, pp. 14-15] to create a kerma factor library.

4) It is recommended to report uncertainties in the displacement or damage energy cross sections due to uncertainties in the nuclear data. The uncertainties should be reported in the form of a variance-covariance matrix. The uncertainty in the nuclear data should be based, if possible, on uncertainty information contained in the ENDF/B-V cross-section library.

5) It is recommended that damage detectors are further developed and that the relationship between measured damage and displacement cross sections (as dpa) is studied.

6) It is recommended to continue the study of competing damage processes (besides dpa) in light water reactors, in fast breeder reactors and in fusion reactor investigations.

   For the calculations of gas production and solid transmutation, accurate excitation functions would be necessary from threshold up to about 30 MeV for (n,γ); (n,xn); (n,tot.H) and (n,tot.He) mostly between 9 and 15 MeV. The list of important materials (e.g., Li, C, N, O, Al, Si, Ti, V, Cr, Mn, Fe, Ni, Cu, Zr, Nb, Mo, Pb) can be found in the IAEA biennual publication WRENDA.

   The two step process ^{58}Ni(n,γ)^{59}Ni(n,α)^{56}Fe contributes considerably to the total helium production in stainless steels at high neutron fluences even in the fast neutron fields. It is recommended that in future work the contribution of this process should be duly accounted for.

   It is further recommended that the Nuclear Data Section encourage measurements of total cross sections up to 40 MeV for the above mentioned reactions. Such measurements are extremely useful for parametrization of nuclear model calculations.

7) The Advisory Group Meeting has noticed that the International Working Group for Reliability of Reactor Pressure Components has (in its Session in Vienna, 4-5 December 1980) agreed to the suggestion of preparing a Status Report of lifetime prediction and surveillance procedures for LWR pressure vessels, for studying the comparability and homogeneity of the procedures (and eventually for making recommendations for improving the homogeneity, by means of a Guidebook).

This plan has found the support of the Advisory Group Meeting.

It is recommended for this purpose to convene a small group of experts of reactor physicists, dosimetrists, and metallurgists.

8) The Advisory Group Meeting urges the Nuclear Data Section to persuade the contributors of WRENDA to realistically redefine the accuracy requirements of their nuclear data needs pertinent to the scope of this meeting.