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LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

NRC METALLURGY AND MATERIALS RESEARCH BRANCH LWR-PV SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM REVIEW GRAPHICS

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I. INTRODUCTION

Aging light water reactor pressure vessels (LWR-PV) are accumulating significant neutron fluence exposures, with consequent changes in their steel 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 U. S. Nuclear Regulatory Commission has established the LWR-PV Surveillance Dosimetry Improvement Program. The primary concern of this program is to improve, standardize, and maintain dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effect of neutron exposure to LWR pressure vessels. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the NRC supported activities at HEDL, ORNL, and NBS and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Jülich, Germany) and several U.K. laboratories have been established.[1] The major benefit of this program will be a significant improvement in the accuracy of the assessment of the remaining safe operating liftime of light water reactor pressure vessels.

A primary objective of the multilaboratory program is to prepare an updated and improved set of dosimetry, damage correlation, and the associated reactor analysis ASTM standards for LWR-PV irradiation surveillance programs, as described in Figures 1-3. Supporting this objective are a series of analytical and experimental validation and calibration studies in "Benchmark Neutron Fields," 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 which are recommended for use in these standards. Consistent and accurate measurement and data analysis techniques and methods, therefore, will have been developed and validated along with guidelines for required neutron field calculations that are used to 1) correlate changes in material properties with the characteristics of the neutron radiation field and 2) predict pressure vessel steel toughness and embrittlement from power reactor surveillance data.

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II. FY 1979 RESEARCH RESULTS - SUMMARY

Relative to the requirements of Appendices G and H of 10 CFR Part 50, neutron induced changes in reactor pressure vessel steel fracture toughness 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:

1. Steel chemical composition and microstructure
2. Steel irradiation temperature
3. Power plant configurations and dimensions - core edge to surveillance to vessel wall positions
4. Core power distribution
5. Reactor operating history
6. Reactor physics computations
7. Selection of neutron exposure units
8. Dosimetry measurements
9. Neutron spectral effects
10. Neutron dose rate effects.

Variables associated with the physcial measurements of PV steel property changes are not considered here and are addressed separately in Appendices G and H of 10 CFR Part 50 and elsewhere.

As older vessels become more highly irradiated, the predictive capability for changes in toughness must improve. Since during the vessel's service life an increasing amount of information will be available from surveillance programs, better procedures to evaluate and use this information can and must be developed. The most appropriate way to make these procedures available is through voluntary consensus standards, such as those now being developed by ASTM Committee E10 on Nuclear Technology and Applications and discussed here.

Important summary highlights of FY 79 research activities of this multilaboratory program are:

A. The preparation of first, revised, or final drafts, Figure 3, of eight of fifteen ASTM standards which focus on the critical neutron exposure (dosimetry), damage analysis (data correlation), and the associated reactor analysis and interpretation aspects, Table 2, of the problem of guaranteeing the safety and integrity of the pressure vessel boundary of LWR power reactors.[2]

B. Initiation and completion of important supporting validation and calibration studies, reviews, and neutron field experimentation, Table 1, which demonstrate and certify the direct applicability of the fifteen ASTM standards (five "practices," five "guides," and five "methods").[3-57]

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C. The completion of key experimental dosimetry/physics studies associated with the PCA low flux version of a PWR pressure vessel mockup, Figures 4-7 and Tables 3-7.[3-9,12,13,20,21,34,37,52,55-57]

D. The initiation of a "Blind Testing" program using the PCA results to assist in the verification of the reliability of neutron physics calculations that are used for predicting PV steel toughness and embrittlement from power reactor surveillance data. The status of this effort is reviewed in a separate paper presented at the NRC 7th WRSR information meeting.[57]

E. The completion of the design and fabrication of the irradiation test assembly, dosimetry, and metallurgy for the PSF experiment, which is a high flux version of a PWR pressure vessel mockup, Figures 8-19 and Tables 8 and 9.[30,32] Irradiation testing in PSF is expected to start in early 1980.

F. The initiation of work associated with the evaluation and reevaluation of exposure units and values for existing and new metallurgical data bases (Reg. Guide 1.99, MPC, EPRI, and others), Figures 20-22 and Tables 10-17.[1,13,14,,19-22,25-29,31,34-39,45-53,55-57]

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OBJECTIVE:

ESTABLISHMENT OF UPDATED AND IMPROVED ASTM STANDARDS FOR LWR PRESSURE VESSEL IRRADIATION SURVEILLANCE DOSIMETRY, DAMAGE CORRELATION, AND ASSOCIATED REACTOR ANALYSIS AND INTERPRETATION PROCEDURES

SUPPORTING ANALYTICAL AND EXPERIMENTAL WORK:

VALIDATION AND CALIBRATION OF THE RECOMMENDED ASTM STANDARDS USING "STANDARD, REFERENCE, AND CONTROLLED ENVIRONMENT BENCHMARK NEUTRON FIELDS," REACTOR "TEST REGIONS," AND OPERATING POWER REACTOR "SURVEILLANCE POSITIONS"

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FIGURE 1. Program Objective and Validation and Calibration.

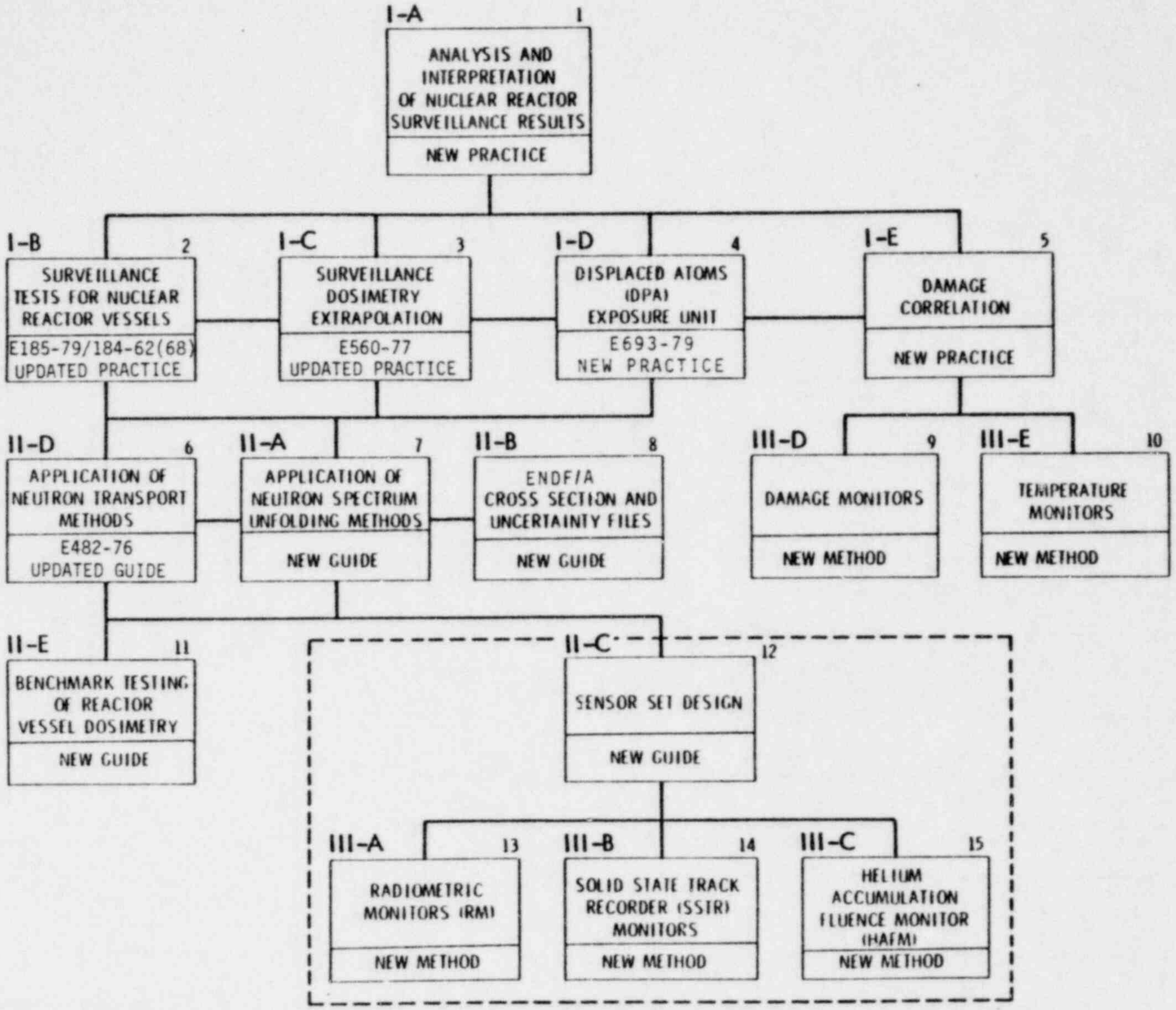

FIGURE 2. ASTM Standards for Surveillance of Nuclear Reactor Pressure Vessels.
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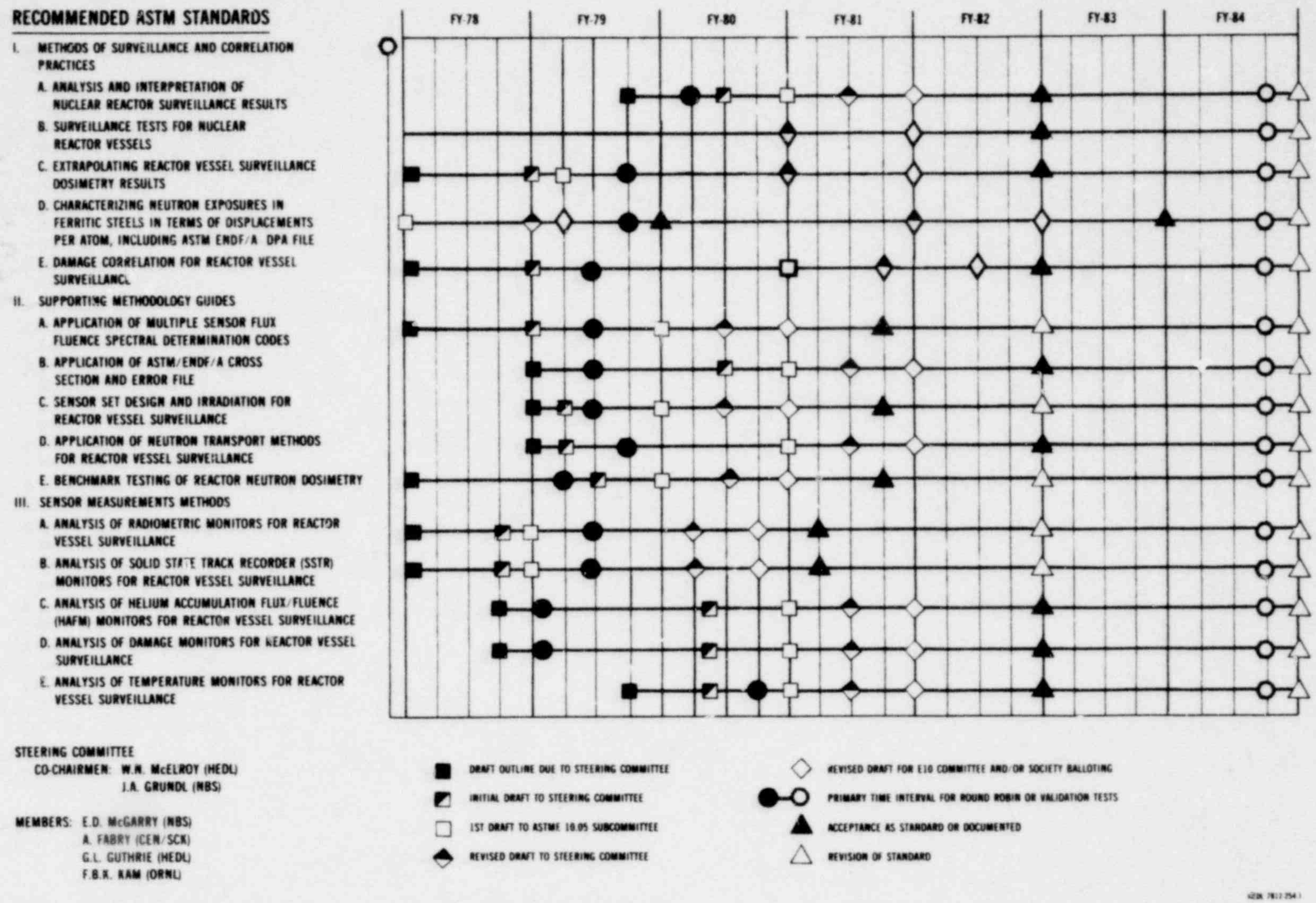

FIGURE 3. Preparation and Validation Schedule for LWR Pressure Vessel Irradiation Surveillance Dosimetry Standards.

III. ASTM STANDARDS FOR SURVEILLANCE OF NUCLEAR REACTOR PRESSURE VESSELS

The status of the preparation and application of the five key standard "Practices," Figure 2, which are associated with "Methods of Surveillance and Correlation" and the associated benefits are discussed in Reference 1.

The new ASTM Master Practice, "Analysis and Interpretation of Nuclear Reactor Surveillance Results," will set out the best available procedures for reactor pressure vessel thoughness and embrittlement predictions. It can be 1) referenced as an instrument of regulation and 2) used for the establishment of improved metallurgical data bases. As shown in Figure 3, this practice will be developed, refined, and tested over the next three years and is expected to be approved as an ASTM Standard. Verification is expected to be completed in an additional two years. The analysis and interpretation steps expected to be contained in the Master Practice are outlined in Table 2.

Two standards (I-C and I-E), Figure 3, dealing with "Methods of Surveillance and Correlation" were completed in draft form. Another, (I-D), which recommends use of an exposure unit that counts the number of neutron-induced displaced atoms in a specimen of LWR-PV steel was accepted as a new standard (E 693-79) for the 1979 Book of ASTM Standards, Part 45. The current schedule for the preparation, acceptance, and revision of all of the new and updated standards is shown in Figure 3.

A summary listing of the type and the status of neutron field studies associated with the validation and calibration of the 15 ASTM Standards is provided in Table 1.

With reference to Table 1, application, testing and the study of neutron dosimetry systems and individual detectors were accomplished for a number of "Benchmark Neutron Fields," research reactor "Test Regions," and PWR and BWR power reactor "In- and Ex-Vessel Surveillance Positions." The results of these dosimetry studies are being combined with available metallurgical data for use in the validation and calibration of the established ASTM Standards and, in particular, the currently applied reactor physics computational tools used for predicting PV steel toughness and embrittlement from test reactor and power reactor surveillance data.

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TABLE 1

LIST OF NEUTRON FIELDS FOR LWR-PV DOSIMETRY VALIDATION AND CALIBRATION STUDIES

Neutron Field	Type of Dosimetry	Status
Benchmarks for Calibration		
• ^{252}Cf and ^{235}U Fission Neutron Irradiation Facilities at NBS (Standard)	Sensor calibrations	Preparation of fluence counting standards in progress for participating laboratories. ^{252}Cf is absolute fast neutron fluence standard.
• ^{235}U Fission Field at CEN/SCK (Standard)	Sensor calibrations	Applied routinely to RM, DM and other sensor calibrations for PCA and other neutron field measurements.
• ISNF Irradiation Facilities at NBS (Standard)	Sensor calibrations (particularly detectors with response range below 0.5 MeV)	Fission rate ratios established with fission chambers. Application to RM and other sensors planned.
• CRMF Irradiation Facility at EG&G (Reference)	Sensor calibrations	Availability being established.
• PCA/PSF Irradiation Facilities at ORNL (Reference)	Sensor validations and surveillance perturbation investigations	Availability being established.
Benchmarks for Transport Calculation Validation		
• Iron Shells Field at CEN/SCK (Reference)	Active and passive RM, SSTR, NE	CEN/SCK spectrometry in progress. NBS fisson chamber measurements completed. Planning stage for passive sensors.
• PCA at ORNL (Reference)	Active and passive RM, SSTR, NE	Radiometric, fission chamber, track recorder, and neutron and gamma spectrometry measurements in progress. Core power distribution established.
Test Regions for Dosimetry Method Validation		
• BSR-HSST Dosimetry-Test (Test Reactor at ORNL)	Radiometric monitors (RM)	Irradiation and dosimetry counting completed. Analysis of Interlaboratory dosimetry started.
• FRJ-1 and FRJ-2 (Test Reactor at KFA, Germany)	RM, HAFM, TM (Melt Wires), DM (Quartz)	"Test Region" irradiations in metallurgical rigs in progress. Interlaboratory dosimetry.
• DIDO, PLUTO, and HERALD (Test Reactors in UK)	RM, HAFM, TM, DM	Planning Stage.
• Buffalo NRL-NRC (Test Reactor at University of Buffalo)	Radiometric monitors	Planning stage.
• University of Virginia (Test Reactor)	Radiometric monitors	Planning stage.
• BR-3 (PWR Power Reactor at CEN/SCK, Mol, Belgium)	RM, HAFM, DM (Quartz), TM (Melt Wires)	In- and Ex-Vessel "Test Region" irradiations in progress.
• Arkansas Power and Light Company PWR Unit #1 (Russelville, Arkansas)	Radiometric monitors	Ex-vessel "Test Region" irradiations and sensor counting in progress.
• Garigliano Reactor (BWR Power Reactor at Rome, Italy)	Radiometric monitors	In-vessel "Test Region" irradiation in progress.
• Brown's Ferry 3 (BWR)	RM, SSTR, Ionization Chambers	In- and ex-vessel "Test Region" irradiations completed.
• McGuire I (PWR)	RM, SSTR, Proton Recoil	Ex-vessel "Test Region" irradiations planned and scheduled.

RM: Radiometric Monitors
SSTR: Solid State Track Recorders
HAFM: Helium Accumulation Fluence Monitors
NE: Nuclear Emulsions
DM: Damage Monitors
TM: Temperature Monitors

TABLE 2

PROCEDURES FOR ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS

Step 1. Establish the basic surveillance test program for each operating power plant. Currently E185-79 is available and is used. However, updated versions of this standard should include the following steps:

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

2. As appropriate, use of measured/calculated DPA damage 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.

Step 2. Establish a reactor physics computational method applicable to the surveillance program. Currently ASTM E482-76 and E560-77 provide general guidance in this area. However, updated versions of these standards should include the following steps:

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

2. 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 useo for verification of this and the previous step.

3. 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.

4. Benchmark validation of the analytical method.

Step 3. Establish methods for relating dosimetry, metallurgy, and temperature data from the surveillance program to current and future reactor vessel conditions. Currently E560-77 provides general guidance in this area. An updated version of this standard should include the following considerations.

1. Improved temperature monitoring.

2. Exposure units to be used to correlate observed changes in upper shelf and RT_{NDT} with neutron environment. This should lead to improved adjustments in trend curves for upper shelf and RT_{NDT}.

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

Step 4. Establish methods to verify steps 2 and 3 and to determine error bounds for the interpretation of the combined results of dosimetry, metallurgical and temperature measurements. Currently, ASTM E185-79 provides general guidance in this area. An updated version of this standard should more completely address the separate and combined accuracy requirements of dosimetry, metallurgy, and temperature measurement techniques.

IV. ORNL Pool Critical Assembly Pressure Vessel Benchmark Facility (ORNL-PCA)

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

Results of studies completed to date indicate that routine LWR power plant calculations of flux, fluence and spectrum, using current S_n transport methods are as accurate as $\pm$ 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.

The PCA Pressure Vessel Benchmark Facility has been fabricated in support of the improvement and validation of the following ASTM Standards:

1. Analysis and Interpretation of Nuclear Reactor Surveillance Results (I-A),
2. Surveillance Tests for Nuclear Reactor Vessels (I-B),
3. Surveillance Dosimetry Extrapolation (I-C),
4. Application of Neutron Transport Methods (II-D),
5. Application of Neutron Spectrum Unfolding Methods, (II-A) and
6. Benchmark Testing of Reactor Vessel Dosimetry (II-E).

Figure 4 shows the overall configuration of the facility. The pressure vessel mockup consists of the thermal shield, the pressure vessel simulator, and the void box with accurately known "plant dimensions."* Access tubes are provided for easy access of dosimetry instrumentation to critical parts of the configuration. Distances can be easily changed to investigate a variety of different configurations. In an x/y configuration, the number x refers to the distance from the core window to the thermal shield and y to the distance between thermal shield and pressure vessel simulator. Figure 5 shows the locations (A1-A8, B1) of access tubes within the pressure vessel mockup.

Extensive core power distribution measurements were carried out in the PCA. A typical comparison of these measurements with transport and diffusion theory calculations is shown in Figure 6. Table 3 summarizes the determination of the absolute core power at a nominal reactor instrumentation reading of 10 kW.

Tables 4 to 7 show preliminary results of the validation for a 40-group, one-dimensional, fixed source, transport calculation. The numbers in these tables are ratios of values obtained from foil dosimetry measurements divided by corresponding values obtained from calculations. In the first two tables, the reaction rates were normalized

*Validation and calibration studies using an existing power plant would be very difficult because there can be uncertainties of the order of 1 inch in the exact dimensions of the vessel size and roundness.

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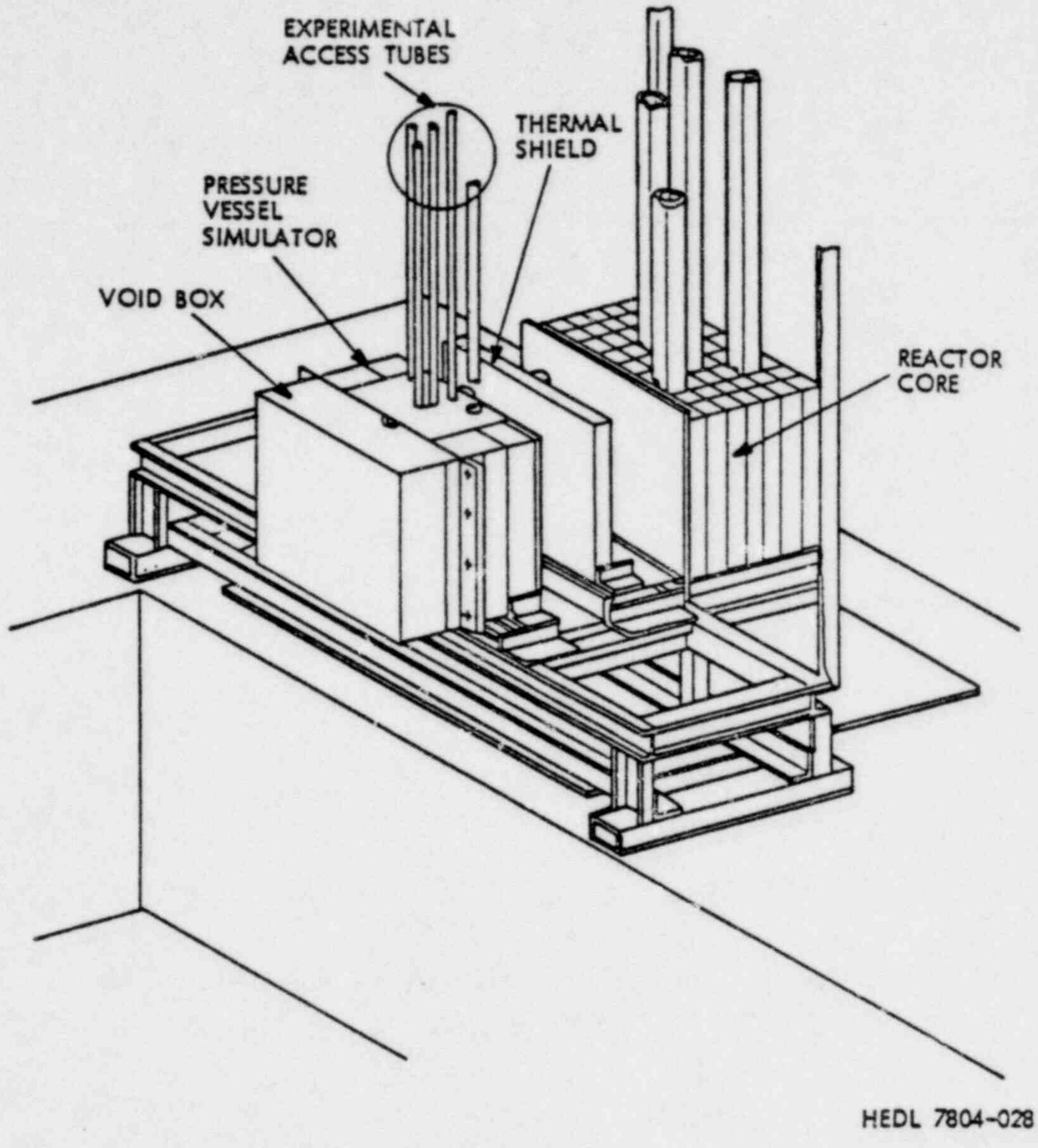

FIGURE 4. Pressure Vessel Wall Mock-up Schematic of Two Equivalent Facilities Constructed at ORNL. The high-flux version at ORR (PSF) will include damage exposure of metallurgical test specimens; the low-flux version near a low-power critical assembly (PCA) will focus on active and passive dosimetry measurements.

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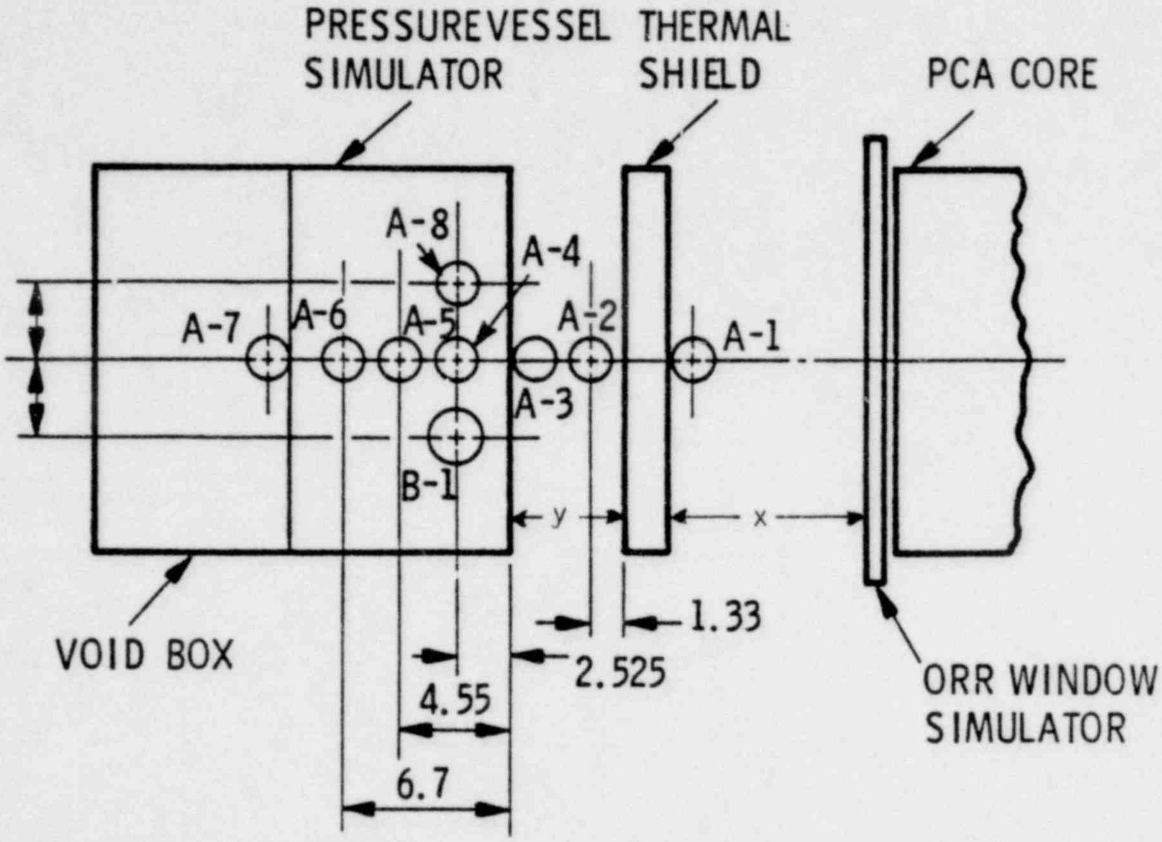

NOTES:

INSIDE DIAM OF HOLES A-1 THROUGH A-8 IS 1.834 IN. (4.658 cm)

INSIDE DIAM OF HOLE B-1 IS 2.469 IN. (6.271 cm)

TUBES A-1, A-2, A-3 ARE REMOVABLE

DIMENSIONS ARE IN INCHES

FIGURE 5. Experimental Configuration for Dosimetry Measurements at PCA.

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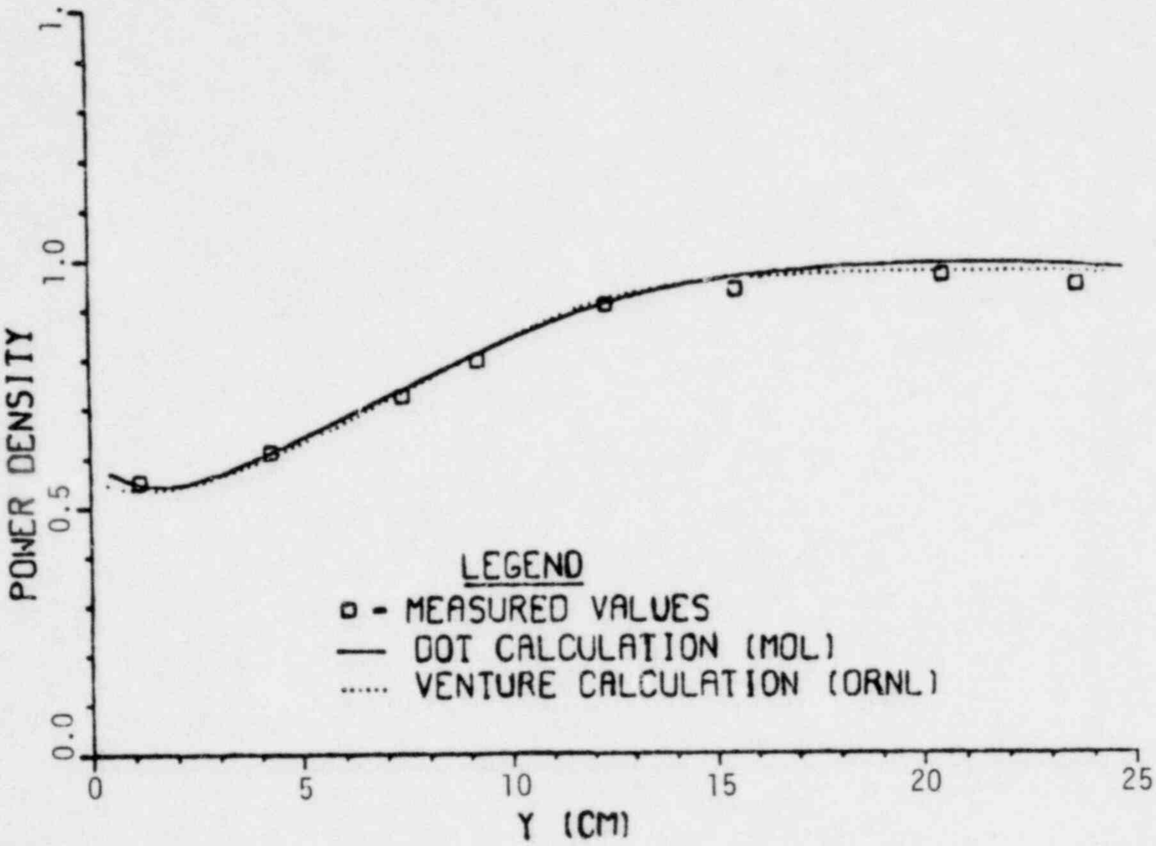

FIGURE 6. Measured and Calculated Radial PCA Power Distributions.

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TABLE 3

ABSOLUTE PCA CORE POWER DETERMINATION AT 10 kW

Position	DT Corrected Count Rate (CPS)	Fission Rate (fission/sec-atom)	Measured Fuel Averaged Power Density (fission/cm^3-sec)	Calculated Power Density	Experimental / Calculated
C5-CC-8	2052.5 $\pm$ 0.3%	1.655×10^{-13}	1.874×10^{7}	0.966	1.940×10^{7}
C5-CC-9	2031.2 $\pm$ 0.5%	1.638×10^{-13}	1.855×10^{7}	0.959	1.934×10^{7}

Effective fission deposit mass = 4.84 ug = $1.24 \times 10^{16} \pm 1.5\%$ (atoms of ^{235}U).

Number density = $1.182 \times 10^{20} \pm 2\%$ (atoms of ^{235}U per cm^3).

Dead Time (DT) = 0.958 $\pm$ 1%.

Average experimental/calculation ratio = $1.939 \times 10^{7} \pm 2.7\%$.

Calculated horizonal power intergral = 852.3 $\pm$ 1.5%.

Calculated vertical power integral = 45.5 $\pm$ 1%.

Total calculated power integral = 38,780 $\pm$ 1.8%.

Total measured power = $7.515 \times 10^{11} \pm 3.3\%$ (fissions/sec).

Watts per fission = $3.204 \times 10^{-11} \pm 2.0\%$.

Power of 25 watt nominal = 24.08 $\pm$ 3.9%.

Measured power ratio 10 kW/25W (nominal) = 405 $\pm$ 1.5%.

Power at 10 kW nominal = 9.75 $\pm$ 4.1%.

relative to the quarter-thickness position in the pressure vessel mockup. The 8/7 and 12/13 configurations are represented in Tables 4 and 5, respectively. In Tables 6 and 7, the reaction rate ratios of a number of foil detectors against that of $^{238}U(n,f)$ are presented. These ratios are used to validate the spectral shape of the calculated spectrum.

Preliminary two-dimensional transport theory calculations are compared to the measurements in Figure 7. This comparison is made in terms of benchmark-field referenced, absolute equivalent fission fluxes normalized to a unit core neutron source strength. It provides an example of the type of result that can be expected in the frame of the PCA transport theory "Blind Test" program.

Similar validation procedures involving proton recoil and $^{6}Li(n,\alpha)$ spectrometry have been carried out in the PCA. These procedures provide the information for the 0.1 to 1.0 MeV energy range which is necessary for an accurate determination of DPA of iron which is now considered a better predictor for radiation damage. Conventional foil dosimetry does not cover this range and it is therefore impossible to determine DPA from foil dosimetry alone. Validation of neutron transport calculations for the 0.1 to 1.0 MeV range in the PCA benchmark field is expected to provide more realistic uncertainty bounds for the prediction of radiation damage in neutron fields where spectrometry cannot be applied.

Absolute gamma spectrometry measurements were carried out in the PCA. The results of this work provide information in support of estimates of 1) gamma heating rates in PSF test assemblies and, 2) photofission corrections for fission dosimeters.

TABLE 4

RATIO OF MEASURED VERSUS CALCULATED REACTION RATES, NORMALIZED TO THE QUARTER-THICKNESS POSITION (A4) 8/7 CONFIGURATION

Position	$^{237}Np(n,f)$)	$^{103}Rh(n,n')$	$^{115}In(n,n')$	$^{238}U(n,f)$
A1 (TSF)*	0.99	1.03	0.83	0.86
A4 (1/4T)	1.0	1.0	1.0	1.0
A5 (1/2T)	1.08	1.11	1.13	1.10
A6 (3/4T)	1.19	1.20	1.23	1.22

*Thermal shield front

TABLE 5

RATIO OF MEASURED VERSUS CALCULATED REACTION RATES, NORMALIZED TO THE QUARTER-THICKNESS POSITION (A4) 12/13 CONFIGURATION

Position	$^{237}Np(n,f)$)	$^{103}Rh(n,n')$	$^{115}In(n,n')$	$^{238}U(n,f)$
A1 (TSF)	0.93	0.85	0.87	0.93
A2 (TSB)*	0.78	0.78	0.76	0.78
A4 (1/4T)	1.0	1.0	1.0	1.0
A5 (1/2T)	1.09	1.11	1.10	1.11
A6 (3/4T)	1.18	1.20	1.17	1.23

*Thermal shield back (surveillance position)

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TABLE 6

RATIO OF MEASURED VERSUS CALCULATED REACTION RATE RATIOS RELATIVE TO $^{238}U(n,f)$, 8/7 CONFIGURATION

Position	$^{237}Np(n,f)$)	$^{103}Rh(n,n')$	$^{115}In(n,n')$	$^{238}U(n,f)$
A1 (TSF)	1.13	0.97	0.95	1.0
A4 (1/4T)	0.99	0.96	1.00	1.0
A5 (1/2T)	0.97	0.96	1.02	1.0
A6 (3/4T)	1.13	0.94	1.00	1.0

TABLE 7

RATIO OF MEASURED VERSUS CALCULATED REACTION RATE RATIOS RELATIVE TO $^{238}U(n,f)$, 12/13 CONFIGURATION

Position	$^{237}Np(n,f)$)	$^{103}Rh(n,n')$	$^{115}In(n,n')$	$^{238}U(n,f)$
A1 (TSF)	1.03	0.92	0.98	1.0
A2 (TSB)	1.05	1.02	1.03	1.0
A4 (1/4T)	1.04	1.00	1.06	1.0
A5 (1/2T)	1.02	1.01	1.05	1.0
A6 (3/4T)	1.00	0.99	1.01	1.0

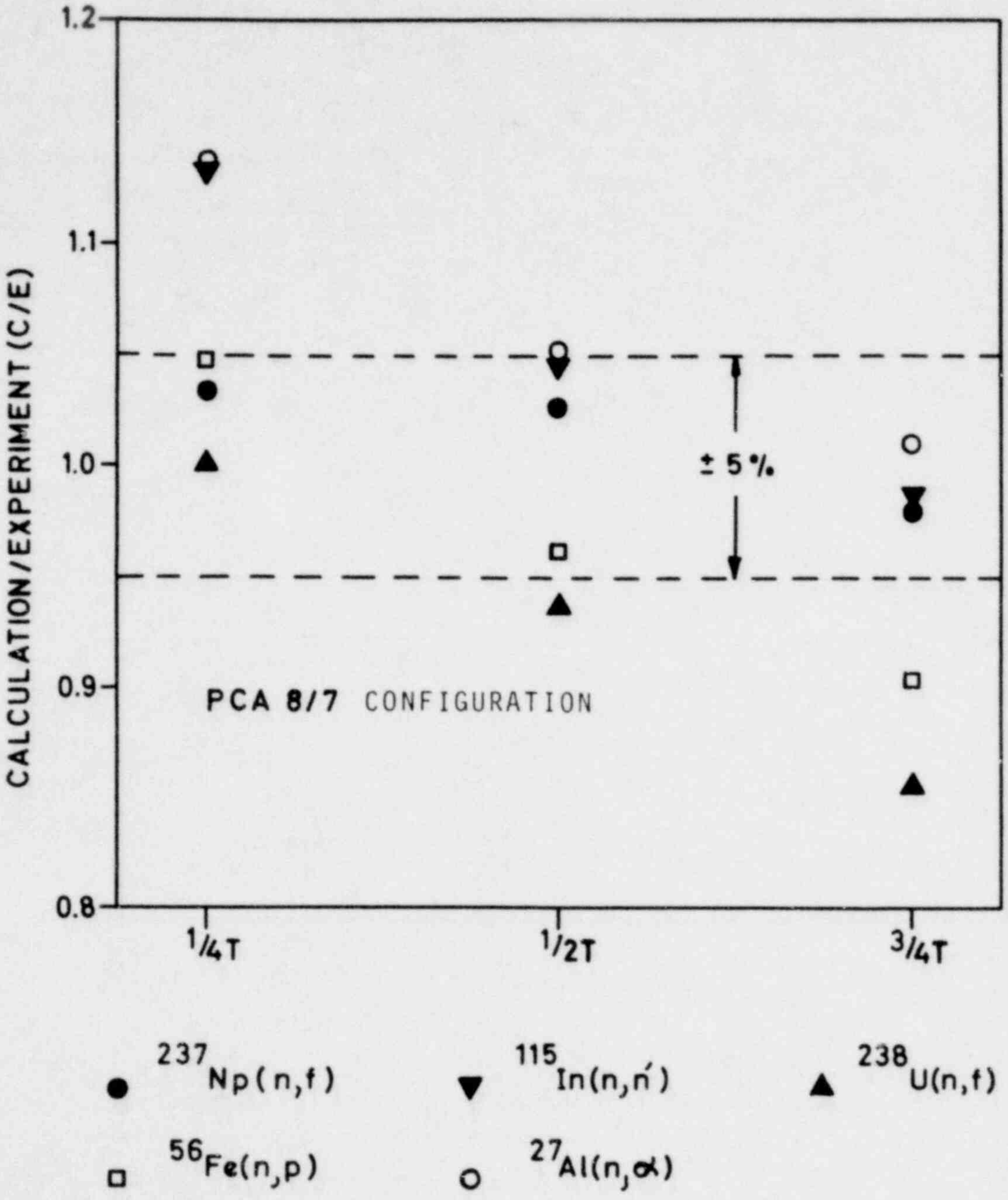


FIGURE 7. Absolute Fast Neutron Fission Fluxes in the PCA Simulated Pressure Vessel Wall for a Unit Core Neutron Source Strength: Comparison Between Calculations and Measurements (Preliminary).

V. LWR Steel Metallurgical Testing in the Pressure Vessel Benchmark Facility (ORR-PSF)

[Variables Studied: 1) Steel Chemical Composition and Microstructure; 2) Steel Irradiation Temperature; 3) Selection of Neutron Exposure Units; 4) Dosimetry Measurements; and 5) Neutron Spectra and Dose Rate Effects.]

Results of studies completed to date indicate that long-term LWR power plant surveillance capsule and short-term test reactor ($\sim$288°C irradiation temperature) neutron induced property change data for steel (base metal, heat affected zone, and weld metal) show significantly different fluence dependencies (power laws of $0.1 \leq n \leq 0.5$). These results should not be combined, therefore, to predict PV steel toughness and embrittlement as a function of neutron exposure without having 1) a more precisely defined and representative data base, 2) a better understanding of the mechanisms causing damage, and 3) tested and verified physical correlation models.

The LWR Metallurgical Pressure Vessel Benchmark Facility (ORR-PSF) is being fabricated in support of the improvement and validation of the following ASTM Standards:

1. Analysis and Interpretation of Nuclear Reactor Surveillance Results (I-A)
2. Surveillance Tests for Nuclear Reactor Vessels (I-B)
3. Surveillance Dosimetry Extrapolation (I-C)
4. Displaced Atom (DPA) Exposure Unit (I-D)
5. Damage Correlation (I-E)

Figure 8 shows the overall configuration of the Oak Ridge Reactor Pool Side Facility (ORR-PSF). Special features for the metallurgical testing will be temperature control, known flux profiles and flux levels, and easy access and flexibility in changing the relative distances between the metallurgical capsules. Figure 9 is an artist's view of the metallurgical capsules which contain the Charpy and tensile specimens, dosimetry capsules, and heating and cooling elements to obtain even and constant temperatures during irradiation.

Fast flux ($E > 1$ MeV) for the 1/4T position in the pressure vessel capsule is estimated to be 3.9×10^{11} n/cm^2-sec. The corresponding fluence value for a two-year irradiation is 2.0×10^{19} n/cm^2. The irradiation conditions have been selected so that information on both flux level and spectral effects should be obtained. Dosimetry and pressure vessel steel materials testing using this new ORR-PSF facility is expected to start in early 1980.

The metallurgical irradiations in the ORR-PSF will take place in several identical packages of specimens, with each package having the geometry shown in Figure 10. Figure 11 shows the geometry for the void box package of specimens. Tables 8 and 9 provide information on the contents of these packages. Figures 12 and 13 show typical loading patterns for space compatible specimens within Charpy holder bars. The number of the packages is now set at six, two to be irradiated sequentially at the first position. The exposure positions are (1) adjacent to the thermal shield, on the side towards the simulated pressure vessel wall (surveillance position), (2) on the front face of the simulated pressure vessel wall, (3) 1/4T position in the PV wall, (4) at the 1/2T position in the PV wall, and (5) an ambient void box capsule position at the rear face of the void box.

Present plans call for the 1/4T position capsule and the number one surveillance position capsule to obtain equal exposure in terms of DPA, with surveillance capsule number two to receive an exposure (DPA) equal to that received by the position two front face capsule.

The planned equality in DPA in the exposure of paired samples exposed in different shaped spectra and at different flux levels will (1) improve the statistical accuracy available in the analysis of the experiment, (2) improve the analyst's ability to draw conclusions about the true shape of the curve of damage per incident neutron vs. neutron energy, and (3) provide information on flux level effects. When combined with other test reactor data, this will ultimately lead to a better understanding of damage mechanisms, to better exposure units, and more confidence in DPA as a correlatable exposure unit, with more accurate future predictions of property degradation.

Figures 14-19 show as-built pictures of HEDL "Backbone," "Gradient," and "Advanced" dosimetry sets and capsules that are identified in Figures 10 and 11. Dosimetry capsule locations for a number of other laboratories are identified in Figures 10 and 11: CEN/SCK, MOL Belgium; KFA, Julich, Germany; AERE, Harwell, England; Combustion Engineering; Babcock and Wilcox; General Electric; and Westinghouse.

The information gained from the PSF dosimetry and metallurgical testing will be used in writing, updating, improving, and validating the procedures recommended in ASTM Standards I-A, I-B, I-C, I-D, and I-E.

1603 077

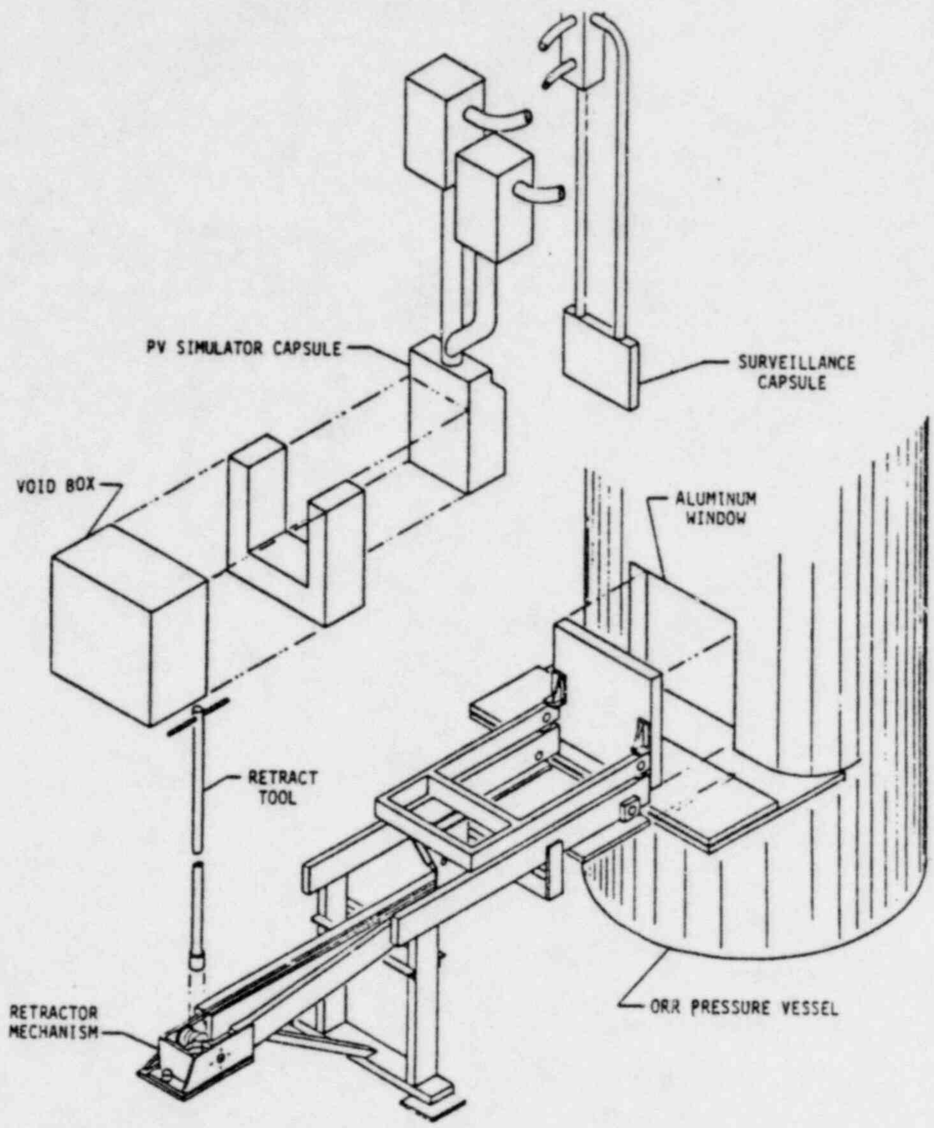


FIGURE 8. LWR Metallurgical Pressure Vessel Benchmark Facility (PSF).

1603 078

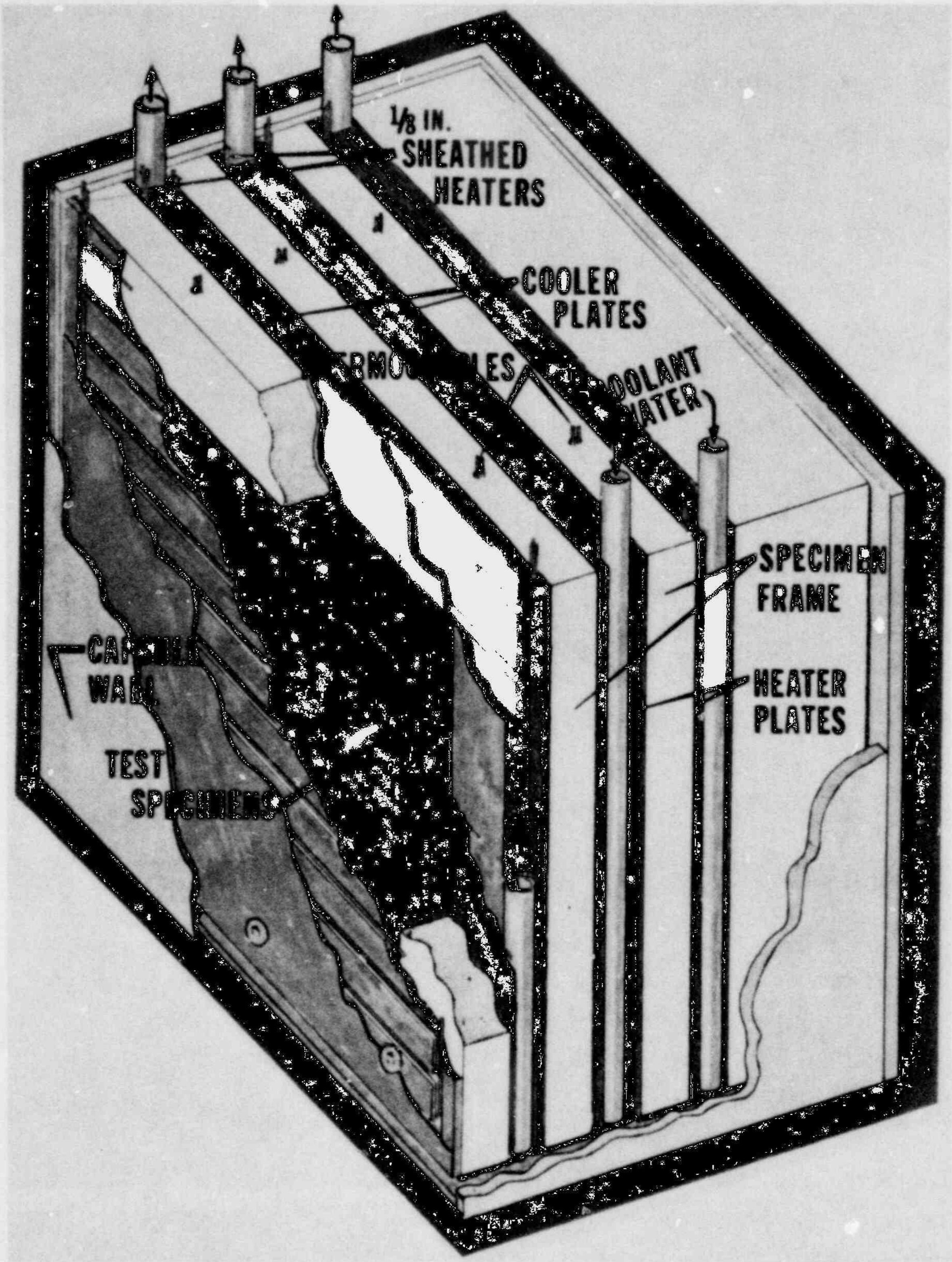
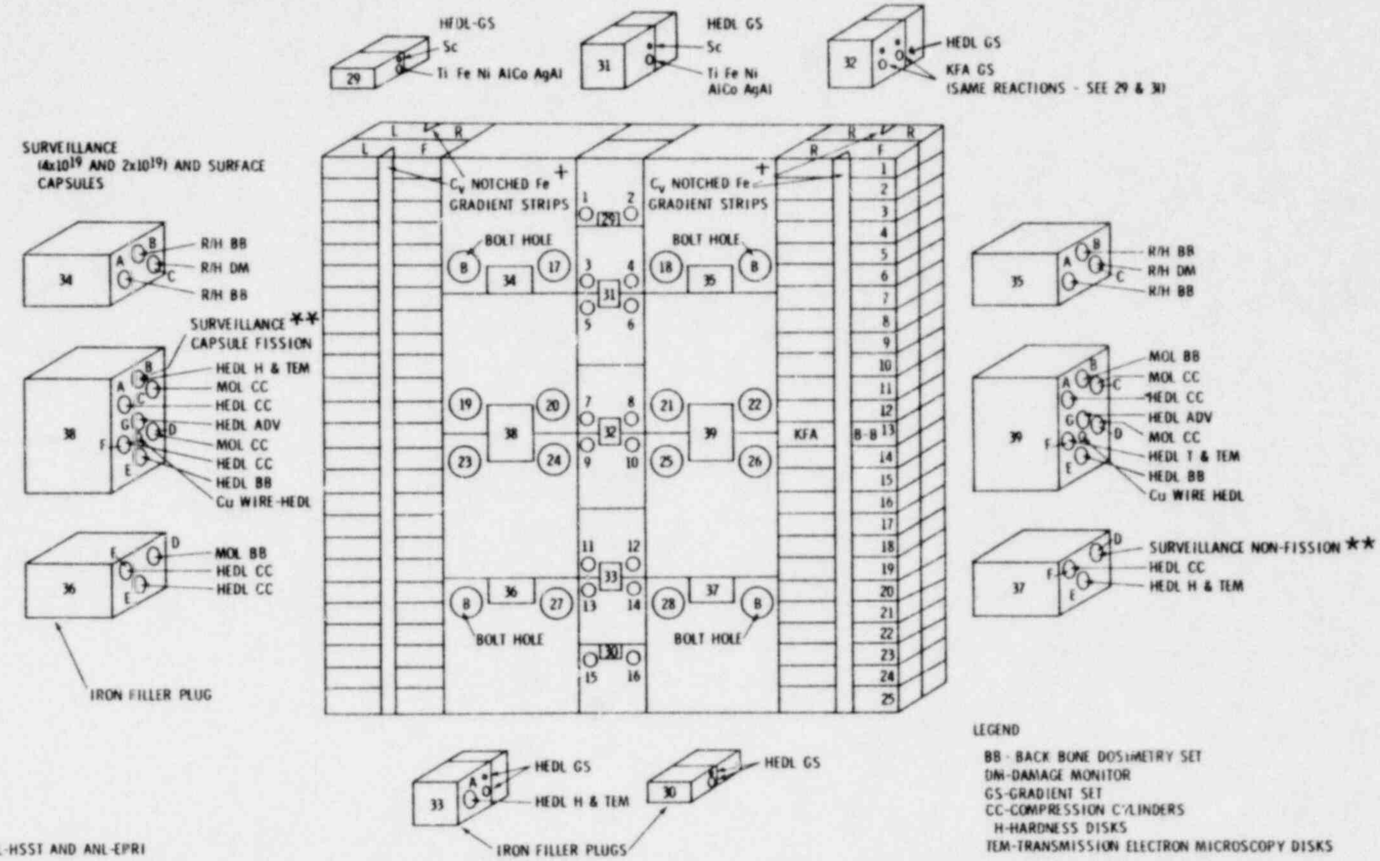


FIGURE 9. PSF Instrumented Irradiation Capsule (Assembled). #7901405-1

[POSITION 1] SURVEILLANCE (~2.0 x 10^{19})* [POSITION 1] SURVEILLANCE (~3.6 x 10^{19})*, AND [POSITION 2] SURFACE (~3.6 x 10^{19})* CAPSULES



SHOULD INCLUDE ORNL-HSST AND ANL-EPRI

PT#	HOLE	LAB	CONTENTS
17	RRA-	HARWELL	CC & TEM
18	"	"	"
19	AVAILABLE FOR UP TO THREE GE DOSIMETRY SETS		
20	" " " " " CE	"	"
21	" " " " " WEST.	"	"
22	" " " " " BW	"	"
23	" " " " " OTHERS*	"	"
24	$4x10^{19}$ SURV. AND SURFACE CAPS.	HEDL	EXPLORATORY SSTR-DM
25	"	"	"
26	AVAILABLE FOR UP TO THREE OTHERS DOSIMETRY SETS		
27		HEDL	H & TEM
28		"	"

PT#	HOLE	LAB	CONTENTS
29		HEDL	GS
30		"	"
31		"	"
32		HEDL & KFA	"
33		HEDL	(GS+(H&TEM IN A)
34	A	RRA/HAR (R/H)	BB
	B	"	BB
	C	"	DM
35	A	"	BB
	B	"	BB
	C	"	DM
36	D	MOL	BB

PT#	HOLE	LAB	CONTENTS
36	E	HEDL	533-CC
	F	HEDL	533-CC
37	D	HEDL	NON-FISSION
	E	HEDL	H&TEM
	F	HEDL	533-CC
38	A	HEDL	302-CC
	B	HEDL	FISSION
	C	MOL	CC
	D	MOL	CC
	E	HEDL	BB
	F	HEDL	302-CC
	G	HEDL	ADV DOS

PT#	HOLE	LAB	CONTENTS
39	A	HEDL	302-CC
	B	MOL	BB
	C	MOL	CC
	D	MOL	CC
	E	HEDL	BB
	F	HEDL	H&TEM
	G	HEDL	ADV DOS
CVRF-13		KFA	BB
CVLF Fe		NOTCHED △	
CVLR Fe		" "	
CVRF "		" "	
CVRR "		" "	

HEDL 7906-170.1

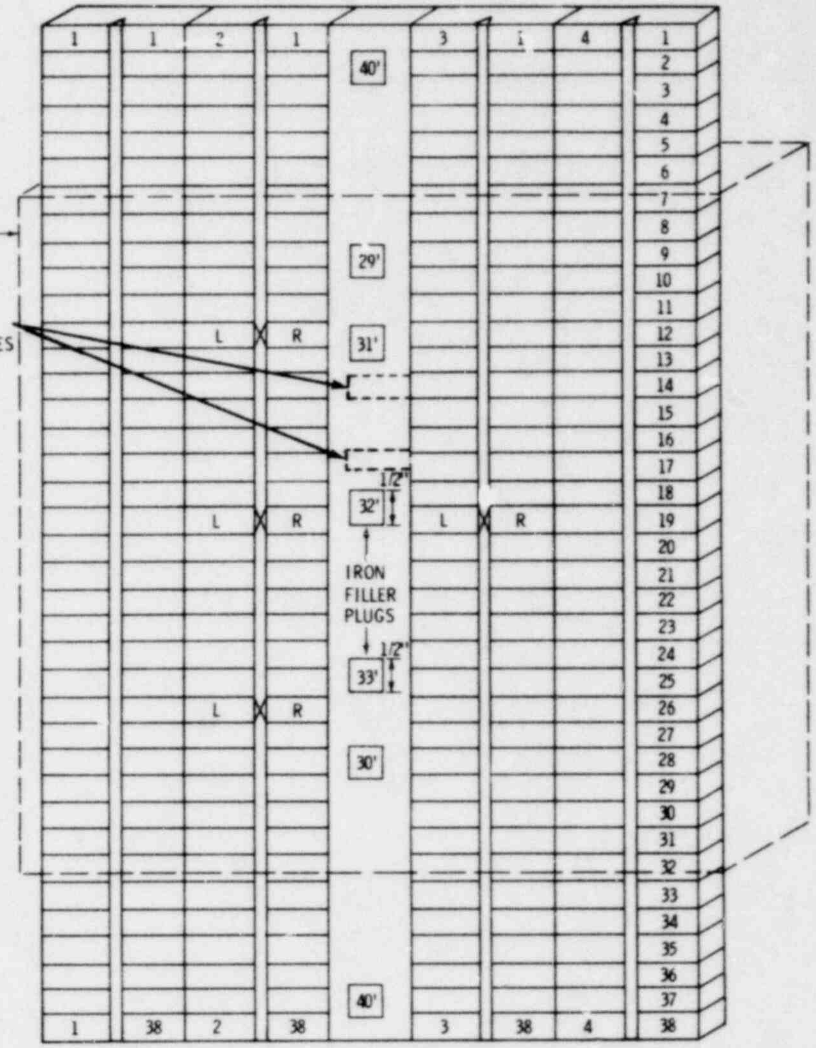
* FLUENCE (n/cm^2)

** Only in Position 1 Surveillance (~2.0x10^{19}) Capsule.

\+ Fabricated by ORNL ~0.010" Dia. Fe will be used instead of notched gradient strips.

FIGURE 10. Metallurgical Specimen Configuration.

1603 080



DOSIMETRY LOCATION

PART LOCATION	LAB	CONTENTS	REMARKS
C_V 2-12 L	HEDL	BACK BONE DOSIMETRY	IN DRILLED C_V 0.257" DIAMETER HOLE
C_V 2-12 R	MOL	BACK BONE DOSIMETRY	IN DRILLED C_V 0.257" DIAMETER HOLE
C_V 2-19 L	HEDL	BACK BONE DOSIMETRY	IN DRILLED C_V 0.257" DIAMETER HOLE
C_V 2-19 R	MOL	BACK BONE DOSIMETRY	IN DRILLED C_V 0.257" DIAMETER HOLE
C_V 2-26 L	HEDL	BACK BONE DOSIMETRY	IN DRILLED C_V 0.257" DIAMETER HOLE
C_V 2-26 R	MOL	BACK BONE DOSIMETRY	IN DRILLED C_V 0.257" DIAMETER HOLE
C_V 3-19 R	HEDL	BACK BONE DOSIMETRY	IN DRILLED C_V 0.257" DIAMETER HOLE
C_V 3-19 L	MOL	BACK BONE DOSIMETRY	IN DRILLED C_V 0.257" DIAMETER HOLE
29'	HEDL	GRADIENT DOSIMETRY	SAME MATERIALS AND APPROPRIATE LOCATIONS AS IN FIGS. 3-1 AND 3-2
31'	HEDL	GRADIENT DOSIMETRY	
32'	HEDL	GRADIENT DOSIMETRY + ADV. DOS. SET 1/4" FROM 31'	
33'	HEDL	GRADIENT DOSIMETRY + ADV. DOS. SET 1/4" FROM 32'	
30'	HEDL	GRADIENT DOSIMETRY	
40' + 41'	HEDL/MOL	GRADIENT DOSIMETRY	AT THE C_V 37 LEVEL
C_V 1 COLUMN	HEDL/ORNL	Fe GRADIENT WIRES	
C_V 2 COLUMN	HEDL/ORNL	Fe GRADIENT WIRES	
C_V 3 COLUMN	HEDL/ORNL	Fe GRADIENT WIRES	
C_V 4 COLUMN	HEDL/ORNL	Fe GRADIENT WIRES	

FIGURE 11. Void Box Rear Surface Capsule. Neg 7911455-2

1603 081

TABLE 8

CHARPY, TENSILE, AND CT SPECIMENS FOR EACH OF FIVE CAPSULES FOR PSF*

Metallurgical Material	Regular** Charpy V	Charpy** Sized Tensile Specimen	0.5TCT	1.0TCT
EPRI Weld Metal	12	2	0	0
NRC/NRL A302B Ref. Plate	18	5	10	5
NRC/NRL A553-03 Ref. Plate	10	2	6	3
KFA/22Ni Mo Cr37, KFA, Jülich, Germany	14	2	0	0
CEN-SCK/COCKERILL SA 508 CLASS-3 FORGING, CEN/SCK, Mol, Belgium	14	2	0	0
To Be Determined, Rolls-Royce and Associates Limited, United Kingdom	12	2	0	0

*Specimens and materials for the void box position are as yet, not completely identified.

**The present accounting of Charpy positions is that 95 Charpy or Charpy equivalent spaces are allotted to participants and five such spaces are available for space compatible specimens, for utilization by assemblers for dosimeters, or for other uses. EPRI has volunteered the use of some EPRI Charpy spaces for space compatible specimens, if needed.

1603 082

TABLE 9

A. SPACE COMPATIBLE SPECIMEN MATERIALS FOR PSF
(ENGINEERING MATERIALS)

Material	Heat Code	Source	Description
A302B Ref. Plate	F23	NRC/NRL	Table 8 Material
A533-03 Ref. Plate	3PT	NRC/NRL	Table 8 Material
Material To Be specified by CEN/SCK	-	CEN/SCK, Mol, Belguim	To Be Specified
A533B-1	1bA	EPRI Archive (HSST-02)	1/4T, 4-1T Compact Halves
A533B-1	1bA	EPRI Archive (HSST-02)	Surface, 6 1/2x10x2 1/4 Piece
A302B	4bA	EPRI Archive (ASTM-CM)	1/4T, 4-1T Compact Halves
SA Weld-Linde 0091 (A533B-1 base)	EP24	EPRI/West/NRL	Hi Cu, Hi Shelf
MMA Weld-E8018-C3 (A533B-1 base)	1mQ	EPRI Archive (CE)	11 x 13 x 6 Piece
A508-2	2bE	EPRI/Archive (B&W)	1/4T, 5 x 9 x 1 Piece
A537-2	7bB	EPRI/Archive (GA)	1/4T, 4-1T Compact Halves

B. SPACE COMPATIBLE SPECIMEN MATERIALS FOR PSF
(MODEL MATERIALS)

Material	Heat Code	Source	Description
A533B-1 (0.03Cu)	N29	NRL/Hawthorne	1/4T, 3 pieces 1/2 x 1/2 x 1/4 in
A533B-1 (0.13Cu)	N27	NRL/Hawthorne	1/4T, 3 pieces 1/2 x 1/2 x 1/4 in
Pure Iron	Fe-6	NRL/Smidt	30 TEM Discs
Fe - 0.3at%V	Fe-2	NRL/Smidt	30 TEM Discs
Fe - 0.3at%Cu	Fe-4	NRL/Smidt	30 TEM Discs
Fe - .34wt%Cu	Alloy 1	West./Spitznagel	Misc. Pieces of Strip & Wire
Fe - 0.1wt%N	Alloy 2	West./Spitznagel	Misc. Pieces of Strip & Wire
Fe - .34wt%Cu - 0.1wt%N	Alloy 3	West./Spitznagel	Misc. Pieces of Strip & Wire

1603 083

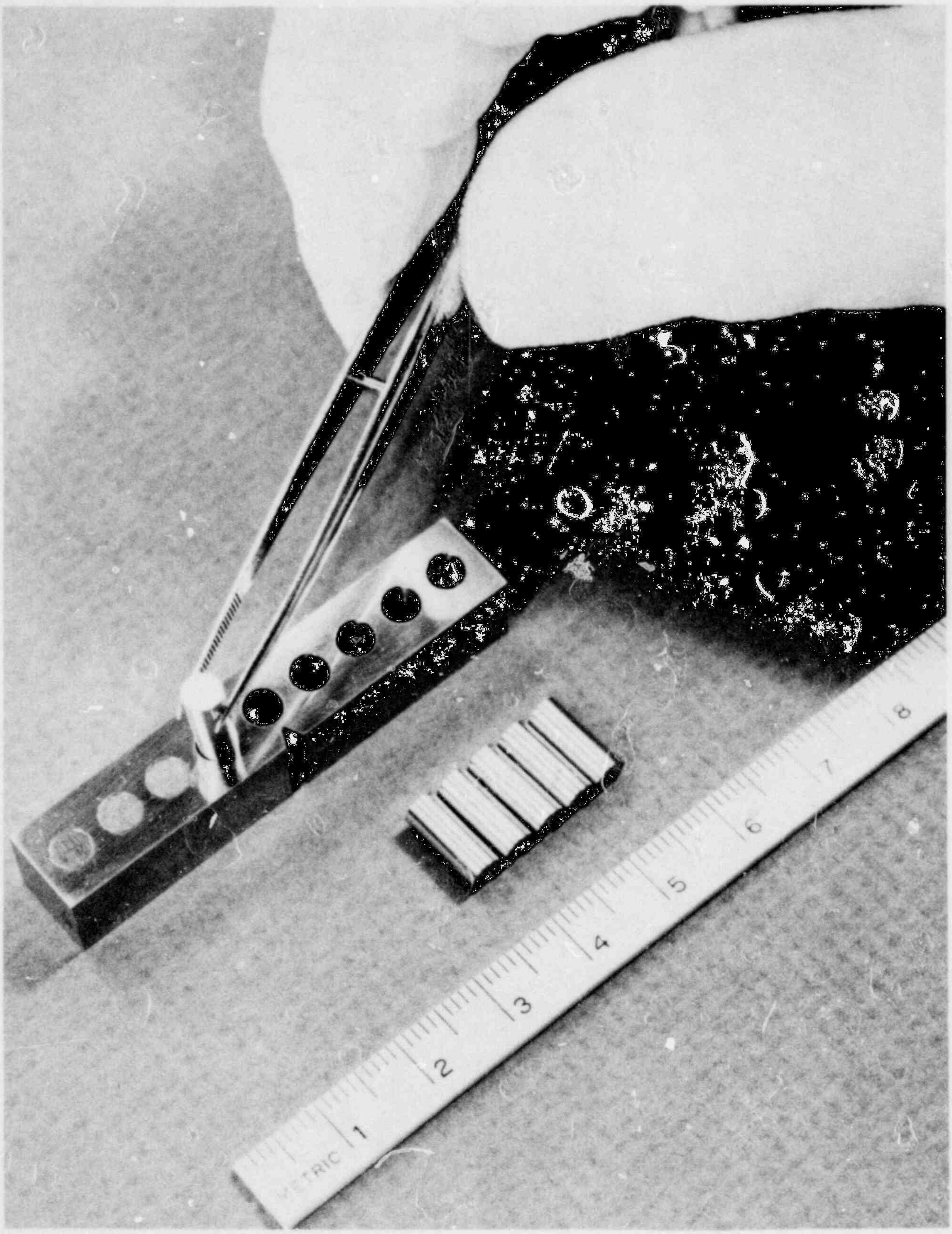


FIGURE 12. Space Compatible Compression Specimens. Neg 7906745-2cn

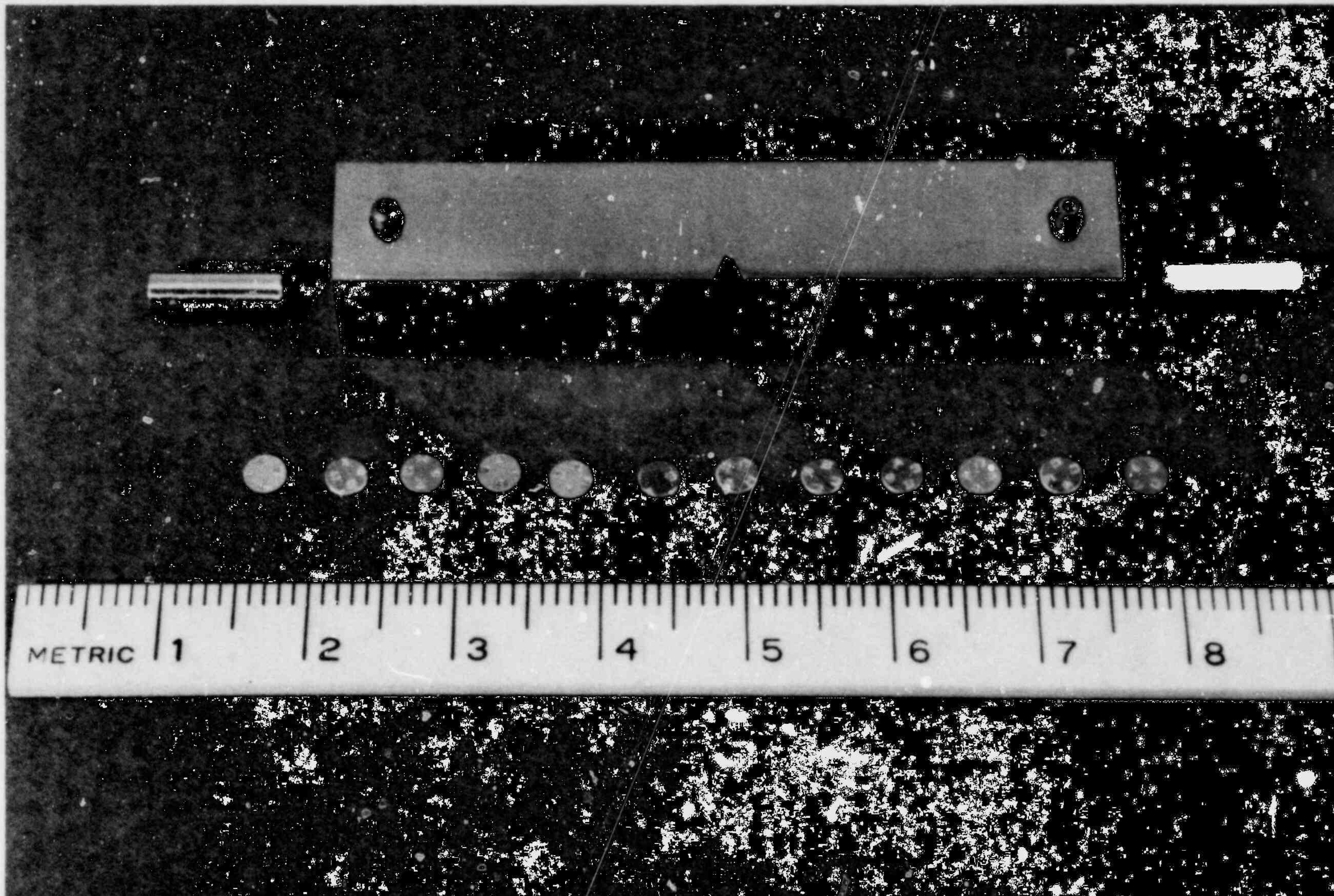


FIGURE 13. Space Compatible Hardness and TEM Specimens. Neg 7906745-6cn

POOR ORIGINAL

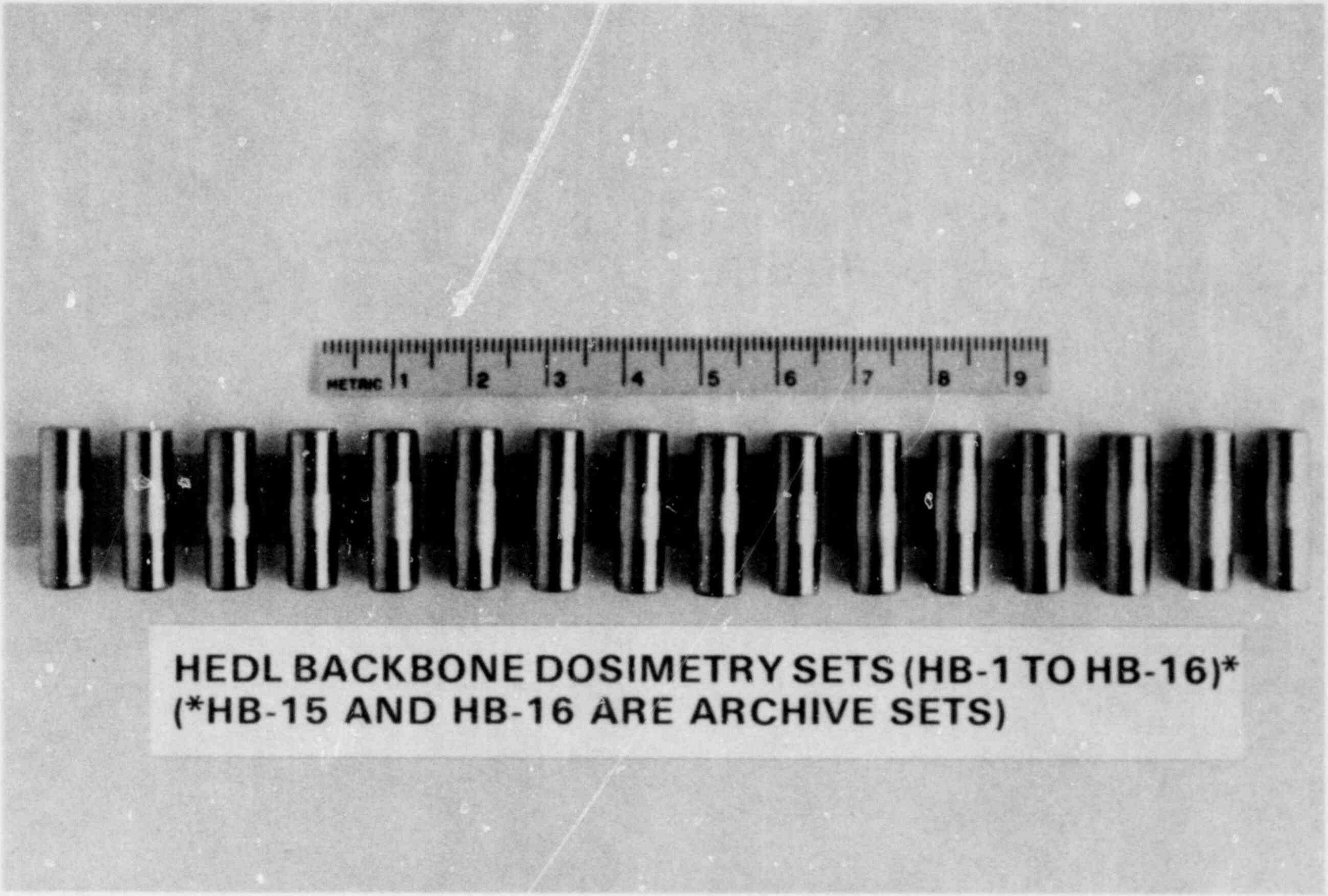


FIGURE 14. HEDL Backbone Dosimetry Sets (HB-1 to HB-16, HB-15 and HB-16 are Archive Sets).
Neg 7909355-5

1603 086

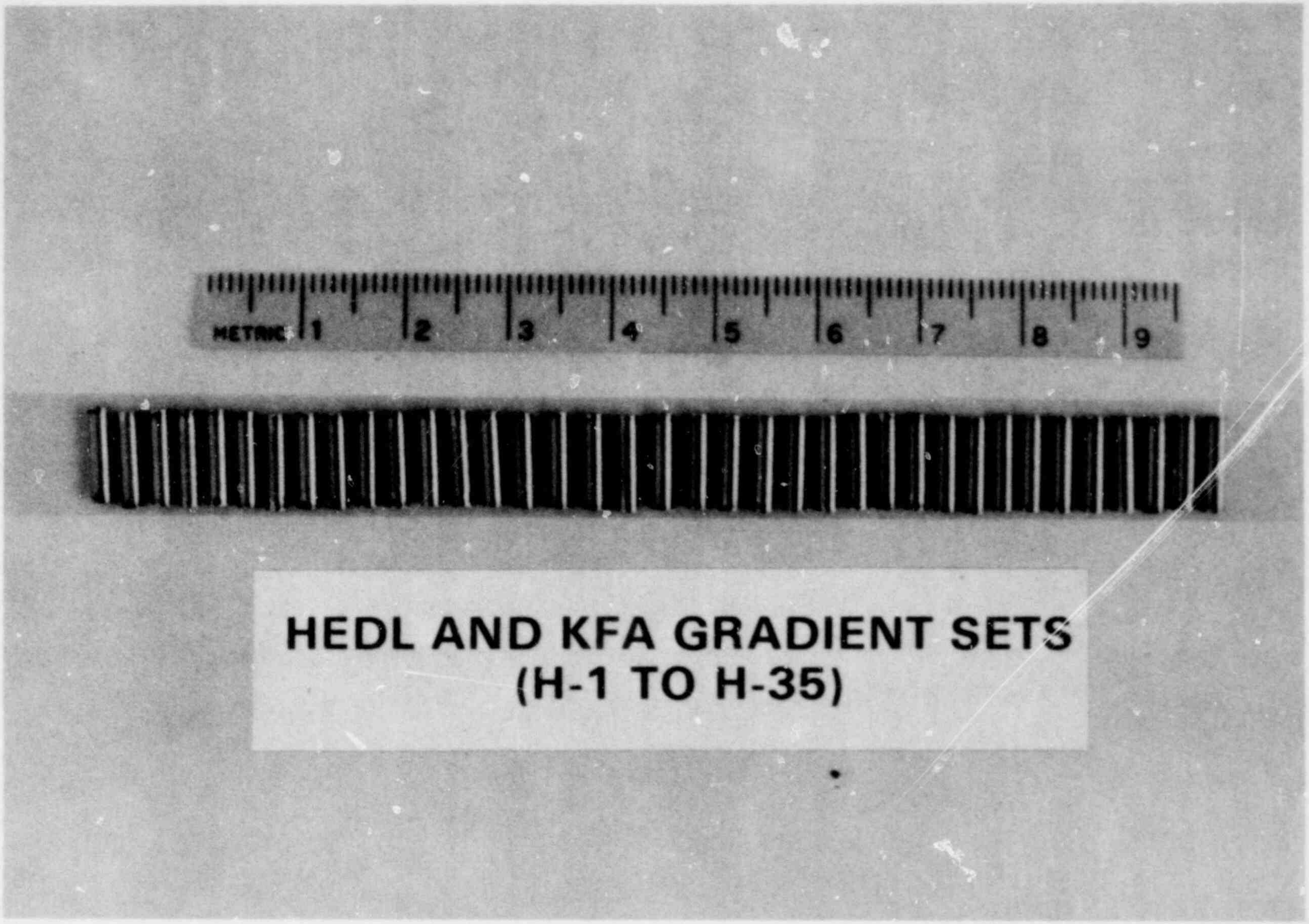


FIGURE 15. HEDL and KFA Gradient Sets. (H-1 to H-35.) Neg 7909355-6

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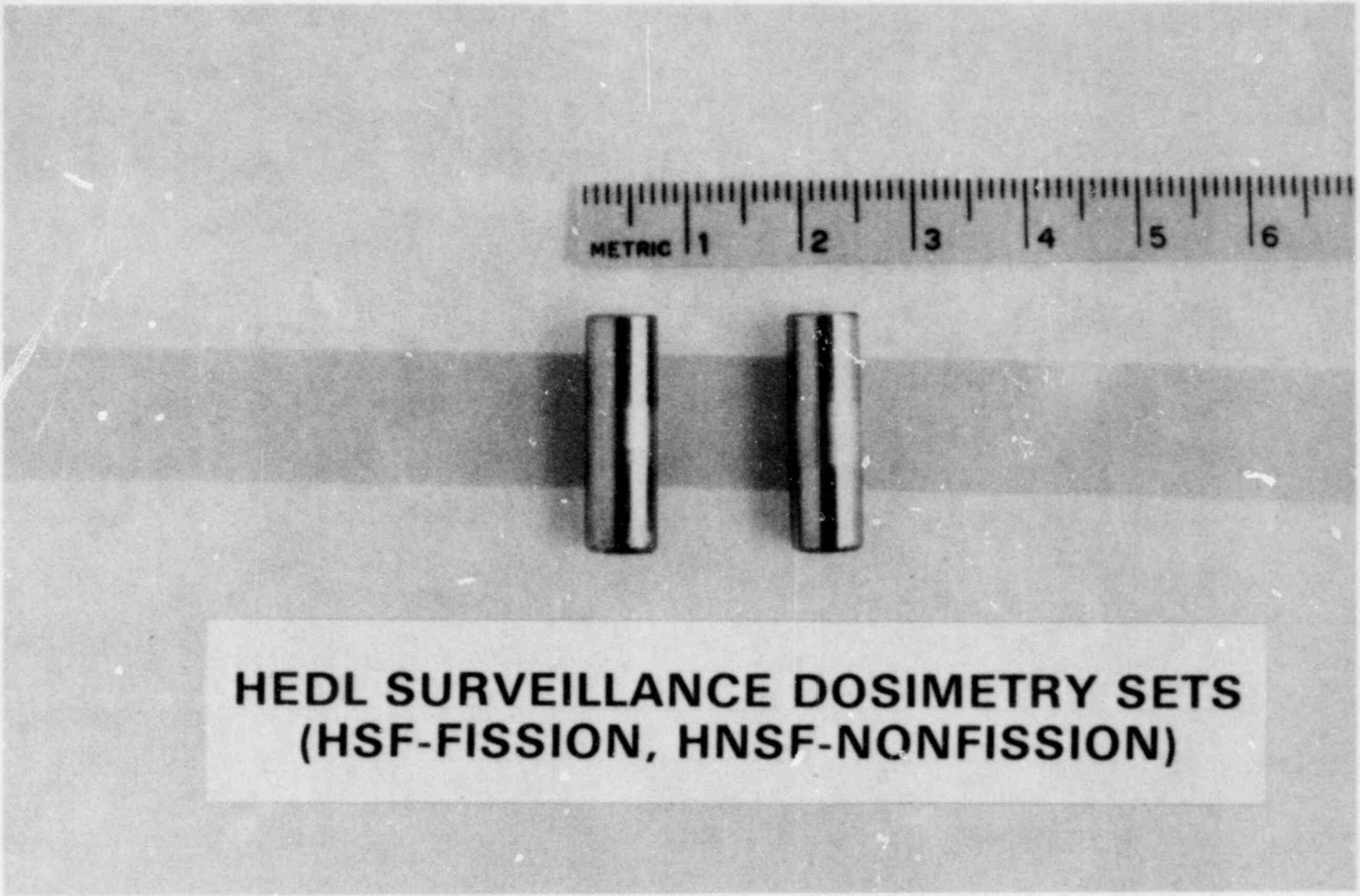


FIGURE 16. HEDL Surveillance Dosimetry Sets (HSF-Fission, HNSF-Nonfission). Neg 7909355-4

1603 088

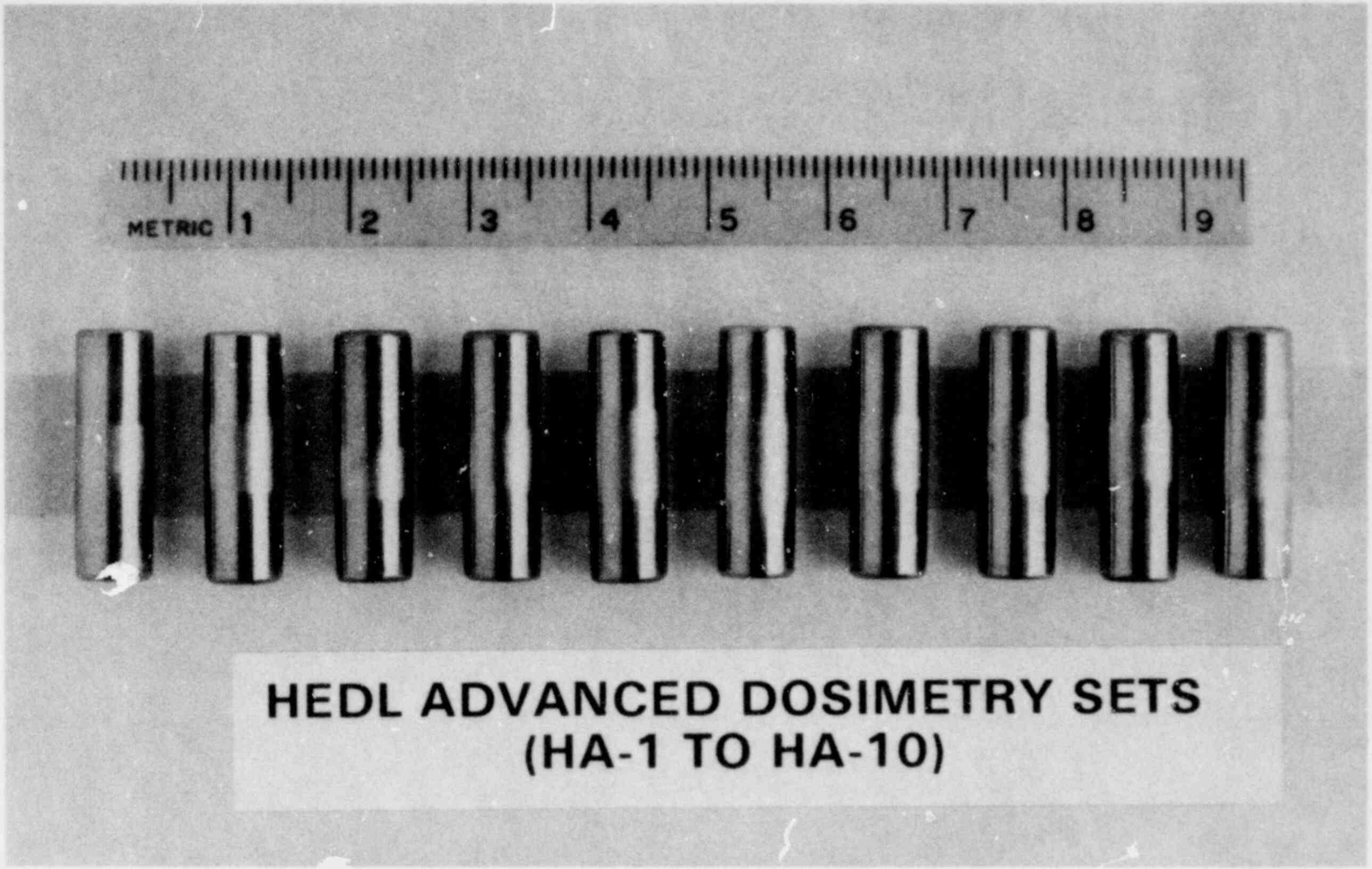


FIGURE 17. HEDL Advanced Dosimetry Sets (HA-1 to HA-10). Neg 7909355-1

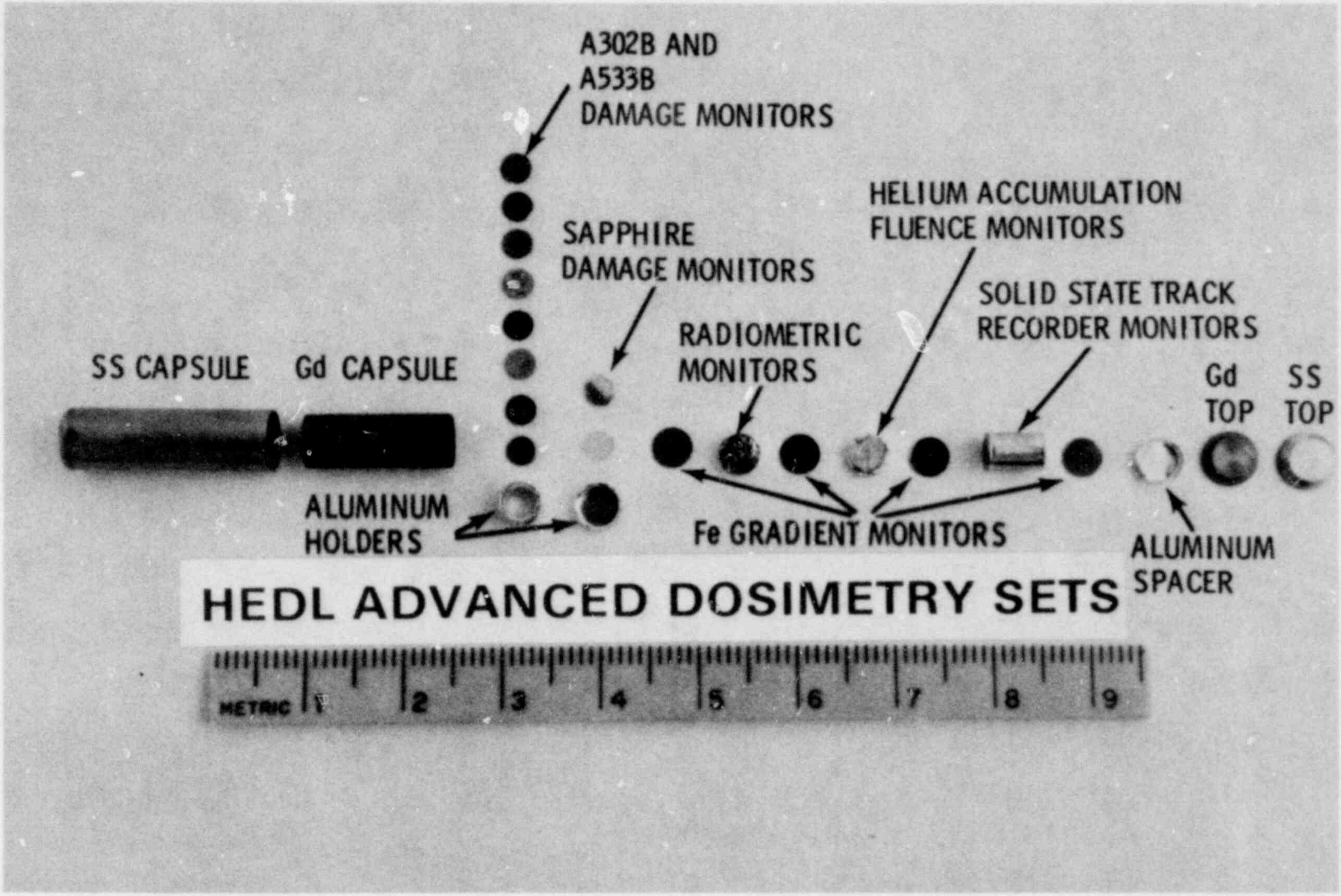


FIGURE 18. HEDL Advanced Dosimetry Sets. Neg 7909355-2

1603 090

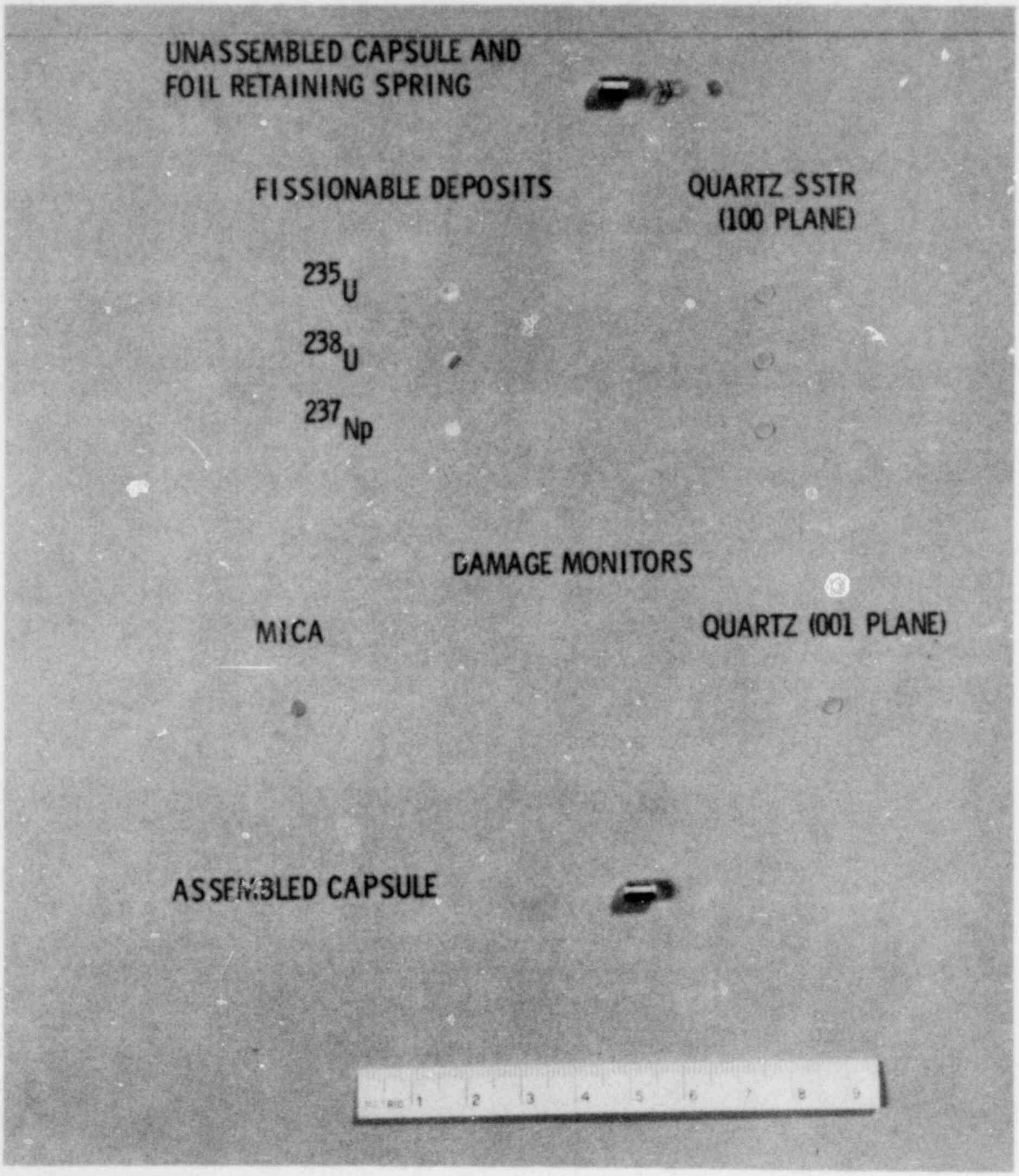


FIGURE 19. Advanced Solid State Track Recorder (SSTR) Capsule for Dosimetry and Neutron Environment Characterization in the Light Water Reactor Metallurgical Pressure Vessel Benchmark Facility (ORR-PSF).

Neg 7909355-3

POOR ORIGINAL

1603 091

VI. Analysis and Interpretation of Research and Power Reactor Test and Surveillance Results

[Variables Studied: All those initially listed in the first paragraph of Section II.]

A primary objective of an EPRI supported program with FCC is to develop a statistically valid radiation embrittlement data base for use in the critical evaluation of the procedures for predicting the fracture toughness of irradiated reactor pressure vessel steels as currently specified in NRC Regulatory Guide 1.99.1.

Analysis of the existing and new additions to the data base (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 analyst," "dosimetry," "metallurgical," and "fracture mechanics" disciplines.

In regard to the chemistry variable, ASTM Standard E185 (I-B, Figure 2) recommends that in addition to Cu and P, consideration should also be given to the interactions of other residual/alloy elements such as Ni, Si, Mn, Mo, Cr, C, S, and V, in estimating the effect of chemical composition. In support of the preparation of the Damage Correlation Standard (I-E), a number of studies have been made at HEDL to find a best expression relating chemistry of pressure vessel steels and the shift in nil ductility temperature which takes place as a result of irradiation. The conclusions which are reached depend on the data set used to develop the model. For weld metal, the best fit found to date has been with an expression of the type

$$\begin{aligned}\Delta NDTT = {} & (A_C \cdot C + A_P \cdot P + A_{Si} \cdot Si + A_{Ni} \cdot Ni + A_{Mo} \cdot Mo + A_{Cu} \cdot Cu + A_V \cdot V + A_{C,Cu} \cdot \\ & C \cdot Cu + A_{P,Cu} \cdot P \cdot Cu + A_{S,Cu} \cdot S \cdot Cu + A_{Si,Cu} \cdot Si \cdot Cu + A_{Ni,Cu} \cdot \\ & Ni \cdot Cu + A_{Mo,Cu} \cdot Mo \cdot Cu + A_{Cu,Cu} \cdot Cu^2 + \text{Constant}) \cdot \\ & (\text{Fluence}/3.0\times10^{19})^{0.28} \qquad (1)\end{aligned}$$

where the A terms are adjustable coefficients and the element symbols represent Wt%. This gave a fit having a one sigma discrepancy between the data and the formula with a value of 19.65°C. The derived values of the coefficiants are given in Table 10. This fit was found with 47 data points from weld data irradiated at temperatures near 288°C. The data was as reported by Varsik and Byrne in the proceedings of the Ninth ASTM "International Symposium on the Effects of Radiation on Structural Materials," July 11-13, 1978, Richland, Washington.

1603 092

TABLE 10

VALUES OF EQUATION 1 LINEAR COEFFICIENTS*
($^{o}C/n/cm^{2}$)

A_C	-0.4693×10^3	$A_{C,Cu}$	0.4545×10^4
A_P	-0.4047×10^4	$A_{P,Cu}$	0.1876×10^5
A_{Si}	-0.1052×10^3	$A_{S,Cu}$	-0.1099×10^5
A_{Ni}	0.2955×10^2	$A_{Si,Cu}$	0.1021×10^4
A_{Mo}	0.5095×10^2	$A_{Ni,Cu}$	0.2005×10^3
A_{Cu}	0.5307×10^3	$A_{Mo,Cu}$	-0.3853×10^3
A_V	-0.4132×10^3	$A_{Cu,Cu}$	-0.2336×10^4
		Constant	0.2872×10^2

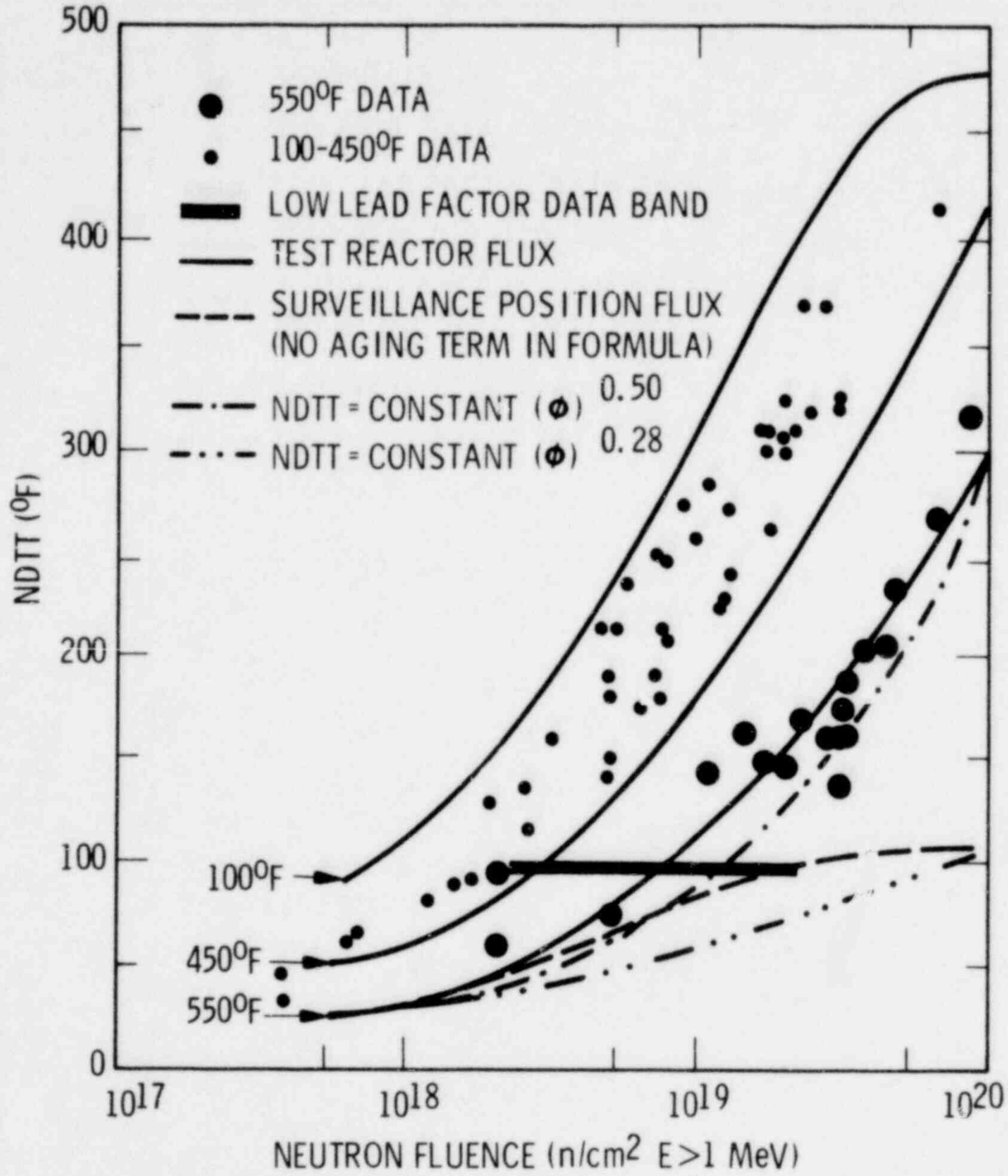
*NOTE: A minus value indicates a beneficial effect, while a positive value indicates a detrimental effect.

Following the decision to use the group of terms, Equation 1, the adopted form was subjected to a least-squares routine which determined not only the best values for the linear coefficients, but also the best value for the exponent in the power law for fluence dependence. This resulted in the choice of 0.28 for the exponent as opposed to the value of 0.43 used by Varsik and Byrne. This gives a result that supports a stronger saturation effect than the usual square root dependence. The 0.28 fluence dependence is compared in Figure 20 with a 0.5 dependence and the results of recent UCSB/FCC modeling studies and Westinghouse surveillance capsule measurements that show a saturation effect. In order to improve the overall correlation of the data, work has been initiated to repeat the above studies using DPA as the exposure unit instead of fluence greater than 1 MeV.

In establishing an improved data base, careful consideration must be given to the variables associated with the use and accuracy of neutron exposure units and values. The total fluence above 1.0 MeV is currently used as the critical spectral parameter for the prediction of reactor material property change. This is a widely-accepted practice, even though the mechanisms which effect the radiation-induced property change cannot be described in such simple terms. The use of the DPA as a more realistic exposure unit for the measurement of materials property change has been recommended and an ASTM Standard E693-79 is now available for its routine application.

The application of DPA as an exposure unit is illustrated in Figure 21, where the Reg. Guide 1.99 trend curves for different compositions of

1603 093



HEDL 7902-089.3

FIGURE 20. Comparison of Fluence Dependencies for the Shift in NDTT.

Neg 7911456-4

1603 094

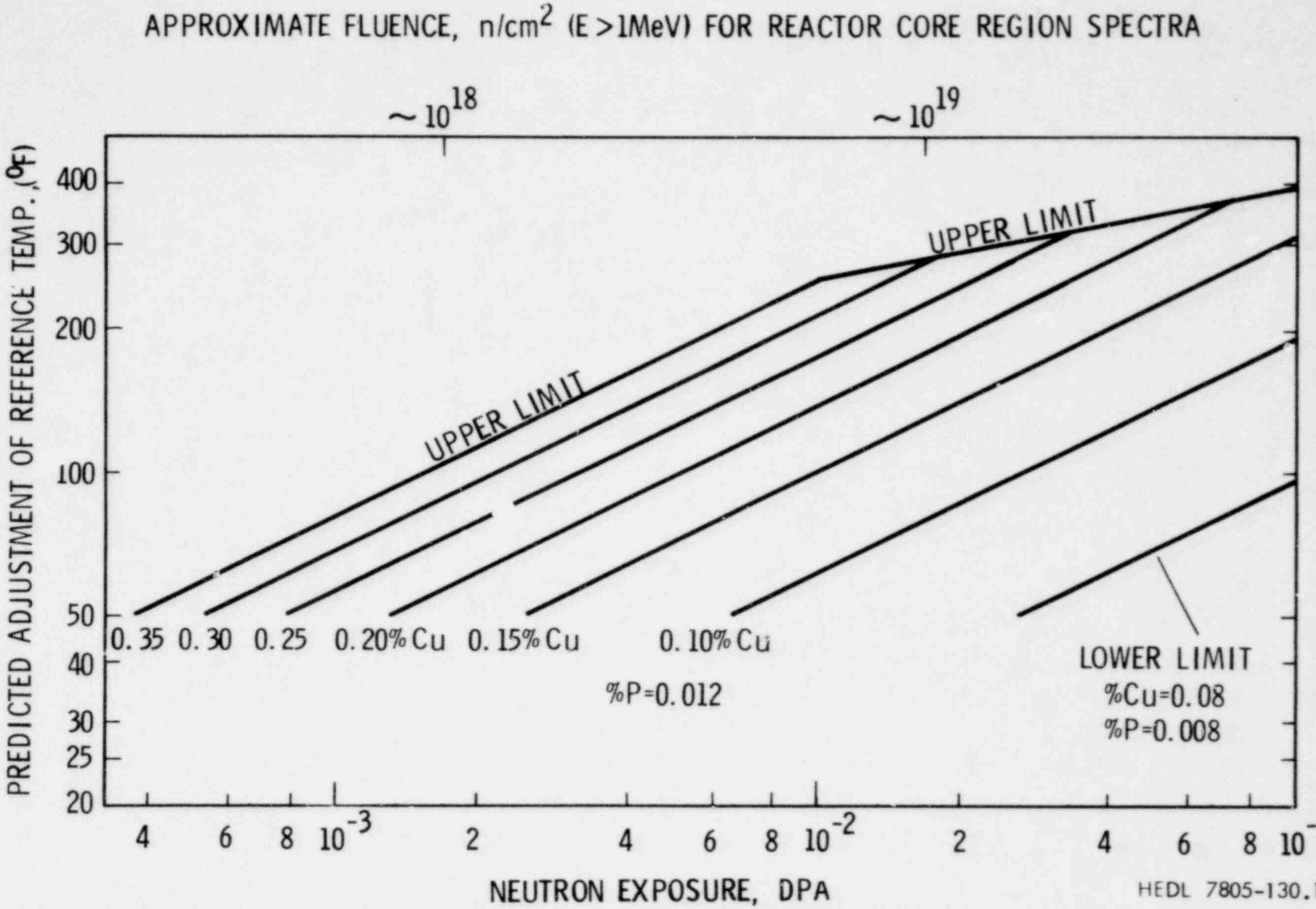


FIGURE 21. Predicted Adjustment of Reference Temperature as a Function of Neutron Exposure and Copper Content. Neg 7911460-1

1603 095

copper and phosphorus for the adjustment of the reference temperature have been replotted using DPA*. In Table 11, the change in DPA exposure shows the significance of the spectrum change from the surveillance through the ex-vessel positions for the 12/13 PCA/PSF configuration. Values for the 1/4 T position in a PWR and the core region for a typical test reactor are also provided. The normalized (parenthesised) values show the relative exposure scale shifts resulting from plotting in units of DPA for a fixed total fluence of 1.0 x 10^{19} n/cm^2 (E>1.0 MeV) at all positions. The effect of using different exposure units to determine a 2-loop PWR power plant surveillance capsule lead factors is shown in Table 12.

The results of the application of both analytical and experimental methods to the determination of the neutron environment internal to PWRs and BWRs are presented in utility surveillance reports and elsewhere. Included is the consideration of the impact of reactor dosimetry set selection, core-vessel geometry, core spatial power distributions, reactor operating history, and capsule perturbation effects. In the validation studies associated with Standard I-A, and the reevaluation of exposure units and values, consideration has been given to previous, updated, and new results for 1) operating power reactor "Surveillance Positions," 2) research reactor "Test Regions," and 3) "Benchmark Neutron Fields." Figure 22 and Tables 13 to 17 provide summary results of these preliminary studies.

Table 13 provides information on the estimated state-of-the-art uncertainties associated with reporting exposure values of A-fluence (E > 1.0 MeV), B-fluence (E > 0.1 MeV), and C-DPA using individual nonfission, fission, or damage monitors. Results are also given for combinations of monitors when "Benchmark Field Referencing" is used. These results make it clear that fission foils or nonfission foils with benchmark field referencing must be used to achieve exposure values with uncertainties near or less than about 30% (2σ). To achieve results with uncertainties much less than this, fission and nonfission monitors with Benchmark Field Referencing are required.

Table 14 provides the results of a brief study of individual and multiple foil reported exposure values for several PWR and one BWR power plant. The results were obtained from the referenced surveillance reports and show the existence of serious discrepancies between fission and nonfission monitor results. In a number of cases the fission foil fluence values are up to a factor of 1.5 to 2.7 higher than the nonfission foil results. These discrepancies are now thought to be largely associated with surveillance capsule flux perturbation effects.

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*A single neutron spectral shape was used for the conversion from ϕ >1 MeV to DPA.

1603 096

TABLE 11

CHANGE IN DPA EXPOSURE FOR A FLUENCE OF 1.0×10^{19} n/cm^2 (E>1.0 MeV)

(In Going from PCA/PSF[a] Surveillance Capsule Location Through the Pressure Vessel Wall into the Ex-Vessel Void Box Cavity)

Position	Fluence (E > 1.0MeV) (n/cm^2)	DPA (DISP/Atom)[c]	$\left[\frac{DPA(<1.0)}{DPA(>1.0)}\right]$[c]	$\left[\frac{\phi(>0.1)}{\phi(>1.0)}\right]$
Surveillance	1.0×10^{19}	.0151 (1.00)[b]	1.29	2.2
Incident PV	1.0×10^{19}	.0158 (1.05)	1.48	2.8
1/4T	1.0×10^{19}	.0178 (1.18)	1.73	3.6
1/2T	1.0×10^{19}	.0205 (1.36)	2.18	5.0
3/4T	1.0×10^{19}	.0243 (1.61)	2.71	6.5
In Cavity	1.0×10^{19}	.0252 (1.67)	2.88	6.9
PWR 1/4T	1.0×10^{19}	.0166 (1.10)	1.72	3.1
Test Reactor Core	1.0×10^{19}	.0141 (0.93)	1.20	1.9

a) For the 12/13 configuration.
b) Normalization point.
c) For these calculations, the DPA cross section below 0.1 MeV has been set equal to zero because of uncertainties associated with the computation of the low energy flux. The actual DPA exposure, therefore, in the PV positions and in the cavity would be somewhat, but not significantly, higher.

TABLE 12

SURVEILLANCE CAPSULE LEAD FACTORS FOR A 2-LOOP PWR POWER PLANT

Position	Lead Factor (E > 1.0 MeV)	(E > 0.1 MeV)	DPA
Surface	3.4 (1.0)*	5.0 (1.0)*	4.0 (1.0)*
1/4T	5.2 (1.5)	5.7 (1.1)	5.4 (1.3)
1/2T	9.1 (2.7)	7.6 (1.5)	8.4 (2.2)

*Ratio to Surface position.

1603 097

TABLE 13

DOSIMETER MONITOR EXPOSURE AND ENERGY RESPONSE MEASUREMENTS AND ASSOCIATED UNCERTAINTIES*

VARIABLE	A. NON-FISSION MONITORS**	B. FISSION MONITORS	C. DAMAGE MONITORS	D. WITH BENCHMARK FIELD REFERENCING***
NEUTRON FIELD	NRC-HSST ORNL-BSR AND SURVEILLANCE TESTS	ARKANSAS P&L, UNIT #1 CAVITY, ORNL-BSR TESTS, AND SURVEILLANCE TESTS	ORNL-PCA/PSF, MOL BR3 TESTS, AND SURVEILLANCE TESTS	SURVEILLANCE TESTS
MEASURED RESPONSE	$>\sim 2$ MeV (Fe, Ni, Ti, Cu)	$>\sim 1$ MeV (^{237}Np, ^{238}U, ^{232}Th)	$>\sim 0.01$ MeV (A302B, A533, QUARTZ, SAPPHIRE)	$>\sim 0.01$ MeV
UNCERTAINTY FLUENCE >1 MeV	$\geq \pm 30\%$	$\leq \pm 30\%$	---	A - $\leq \pm 30\%$ B - $\leq \pm 20\%$ A+B - $\leq \pm 15\%$
UNCERTAINTY FLUENCE >0.1 MeV	$\geq \pm 60\%$	$\geq \pm 40\%$	---	A - $\leq \pm 60\%$ B - $\leq \pm 40\%$ A+B - $\leq \pm 30\%$
UNCERTAINTY DPA	---	---	$\leq \pm 30\%$	A - $\leq \pm 30\%$ B - $\leq \pm 30\%$ A+B - $\leq \pm 30\%$ A+B+C - $\leq \pm 30\%$

*BOUNDS AT THE 95% CONFIDENCE LEVEL

**DOES NOT INCLUDE USE OF Nb

***INCLUDES FISSION, PCA, AND PSF PRESSURE VESSEL MOCKUP NEUTRON FIELDS

1603 098

TABLE 14

REPORTED SURVEILLANCE CAPSULES SINGLE FOIL FLUX/FLUENCE VALUES [ϕ_i > 1 MeV] - RELATIVE TO $^{54}Fe(n,p)$

Reactor Name (Vendor-Type, Country, Operation Date)	Service Laboratory Report	SINGLE FOIL FLUX/FLUENCE >1MeV				
		$\phi_{^{58}Ni(n,p)}$	$\phi_{^{63}Cu(n,\alpha)}$	$\phi_{^{238}U(n,f)}$(e)	$\phi_{^{237}Np(n,f)}$(e)	Reference
Point Beach #1	BMI (1973)	1.09	1.63	1.61	2.17	(1)
(West. PWR, USA, 12/70)		[Reported Surveillance Value: 1.0(Fe)](d)				
Same (Angle A)	WEST.(1979)(a)	0.80	1.51(b)	1.03	1.25	(2)
Same (Angle A + 180°)	WEST.(1979)(a)	1.01	1.14(b)	1.13	1.01	(2)
		[Reported Surveillance Value: 1.0 (Fe)](d)				
Average Values For Seven WEST. Power Plants	WEST.(1979) (a)	0.97	1.25(b)	1.08	1.15	(2)
Humboldt Bay 3	GE (1967)	0.88	0.80	-	-	(3)
(GE BWR, USA, 8/63)		[Reported Surveillance Value: 1.0 (Fe)](d)				
San Onofre #1	SWRI(1971)	1.00	1.27	1.10	1.42 (BNW Spectrum 1)	(4)
(West. PWR, USA, 1/68)	SWRI(1971)	1.05	0.88	1.29	1.45 (BNW Spectrum 2)	(4)
		[Reported Surveillance Value: 0.85 (SAND II, multiple foils)](d)				
OCONEE #1	B&W (1975)	1.18	-	2.50	2.70	(5)
(B&W PWR, USA, 7/73)		[Reported Surviellance Value: $\frac{1.0 + 2.5}{2}$ = 1.76 (Fe+U)](d)				
DOEL I	CEN/SCK (1979)	1.09	1.51(b)	-	2.41	(6)
Belgium-West. PWR Belgium 1/75)	CEN/SCK (1979)	1.09	1.06(c)	-	2.41	(6)
		[Reported Surveillance Value: ~ 1.09 (Ni)](d)				

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TABLE 14 FOOTNOTES

(a) Surveillance capsule flux perturbation corrections were calculated by Westinghouse to provide necessary correlations between the U, Np, Cu, Ni, and Fe results. None of the other results shown in the table have been corrected for perturbation effects.

(b) ENDF/B-IV $\sigma(E)$ for $^{63}Cu(n,\alpha)$

(c) Mann-Schenter $\sigma(E)$ for $^{63}Cu(n,\alpha)$

(d) These reported surveillance capsule measured fluence values are used for correlating the surveillance capsule metallurgical data with other test and power reactor data. They are also used for making localized predictions of expected pressure vessel lifetime neutron exposures and/or can be used to simply confirm the correctness of one, two, and three dimensional reactor physics computations.

(e) Based on ^{137}Cs analysis.

NOTE: The Ni, Fe, and Cu provide experimental fluence data for time periods up to about 1 year, 5 years, and 25 years with a knowledge of the surveillance capsule flux level time history, while this information is not needed for a reliable interpretation of fission foil $^{137}Cs(t_{1/2} \sim 31$ years) results.

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For Point Beach #1, the 1979 Westinghouse reanalysis, which included the calculation of perturbation corrections, shows good agreement for the combined Fe, Ni, U^{238}, and Np^{237} results. For San Onofre #1, the Southwest Research Institute results using the SAND II adjustment code* show much better consistency than the Point Beach #1 (1973 BMI) and Oconee #1 (1975 B&W) results. These BMI and B&W results did not consider surveillance capsule perturbation effects. If a reasonably good reactor physics input spectrum is available, together with a set of accurately measured reaction rates (about $\pm$ 5%, 1σ), adjustment codes can modify the input spectrum to properly account for perturbation effects. The accuracy with which this can be done, however, depends on the combined energy response of the selected set of foil monitors. For fluence > 1.0 MeV, Np^{237}, U^{238}, and Fe^{54} provide a good combination of monitors; see Table 13. It should be noted here that the Westinghouse fission foil results, Table 14, have also been corrected for photofission effects, while those of the other servicing laboratories have not.** The Belgian CEN/SCK laboratory has recently obtained much more consistent dosimetry results for the fission and nonfission foils for the DOEL I second surveillance capsule. The Table 14 results for the DOEL I are for the first surveillance capsule. In the analysis for the second capsule, they made use of Westinghouse calculated capsule perturbation correction factors.

The following example demonstrates the improved accuracy that can be achieved by benchmark referencing dosimetry measurements made in an applied neutron field. The bases of benchmark field referencing are irradiations performed for the purpose of detector calibrations in a neutron field of known energy spectrum and intensity. Then measurements in the applied field can simply be related to the relative responses of the detectors in the two fields. The applied field in this example is the air-filled cavity external to the reactor pressure vessel of Arkansas Power and Light Company's Unit #1 PWR, hereafter designated ANO-1. Two experimental capsules containing dosimeters representative of those used for in-vessel surveillance dosimetry were hung at mid-core elevation for a 55-day operations cycle, 98% of which was at full power, and for which all power history is known. The dosimeter-foil counting was accomplished to better than 5% by HEDL. Analysis of the data was accomplished by NBS.

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*With an input spectrum based on less accurate reactor physic computations than those performed by Westinghouse.

**Laboratory-to-laboratory corrections for the use of different cross sections and fission yields have not been considered, but will be in future reanalysis of these data.

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Briefly, the conventional method of using total-spectrum averaged cross sections to derive the fast neutron flux (greater than 1 MeV) from measured reaction rates was compared to analyses of the same data after it was referenced back to benchmark neutron fields. Tables 15 and 16 compare the precision of the two methods. By conventional analysis, the mean fast fluence (flux-time product) from four different nuclear reactions is $1.55 \times 10^{15} n/cm^2 \pm 19\%$ (3 sigma). After referencing to benchmark fields, the standard deviation is reduced to 7.2%, a factor of three improvement. Table 17 provides the best value of the full power flux in the ex-vessel cavity of ANO-1. This result is derived from the average of the benchmarked fluence results and the detailed power history.

Figure 22 is a few-group display of the spectrum, the average number of displacements per atom of steel (DPA), and the nuclear reactions used for dosimetry in the ANO-1 Cavity. For each quantity, the figure indicates the energy range over which 5% to 95% of the response (or in one case, the spectrum) occurs. This range is further subdivided by a caret, to indicate the energies corresponding to 50% response, and two vertical lines, to indicate the energies corresponding to 25% and 75% response. It is important to observe from Figure 22 just how selective various nuclear reactions can be in measuring portions of the neutron energy spectrum. This is a very desirable feature for spectroscopy but not necessarily desirable for a total flux monitor. The only two detector reactions shown which monitor most of the neutrons in the spectrum are the fast fission reactions in U^{238} and Np^{237}. For DPA, the choice is more limited to the Np^{237}. Although Np^{237} was not included in the ANO-1 cavity experiment, it should have been.* Its importance as a total fast fluence monitor strongly suggests its inclusion in the basic dosimeter set for all power reactor pressure vessel surveillance programs.

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*A suitable Np^{237} foil was not available at the time of this irradiation. Subsequent repeat experiments to measure reproducibility and relative magnitudes of flux at other cavity locations have been performed with a Np^{237} dosimeter included and their results will be reported as data become available.

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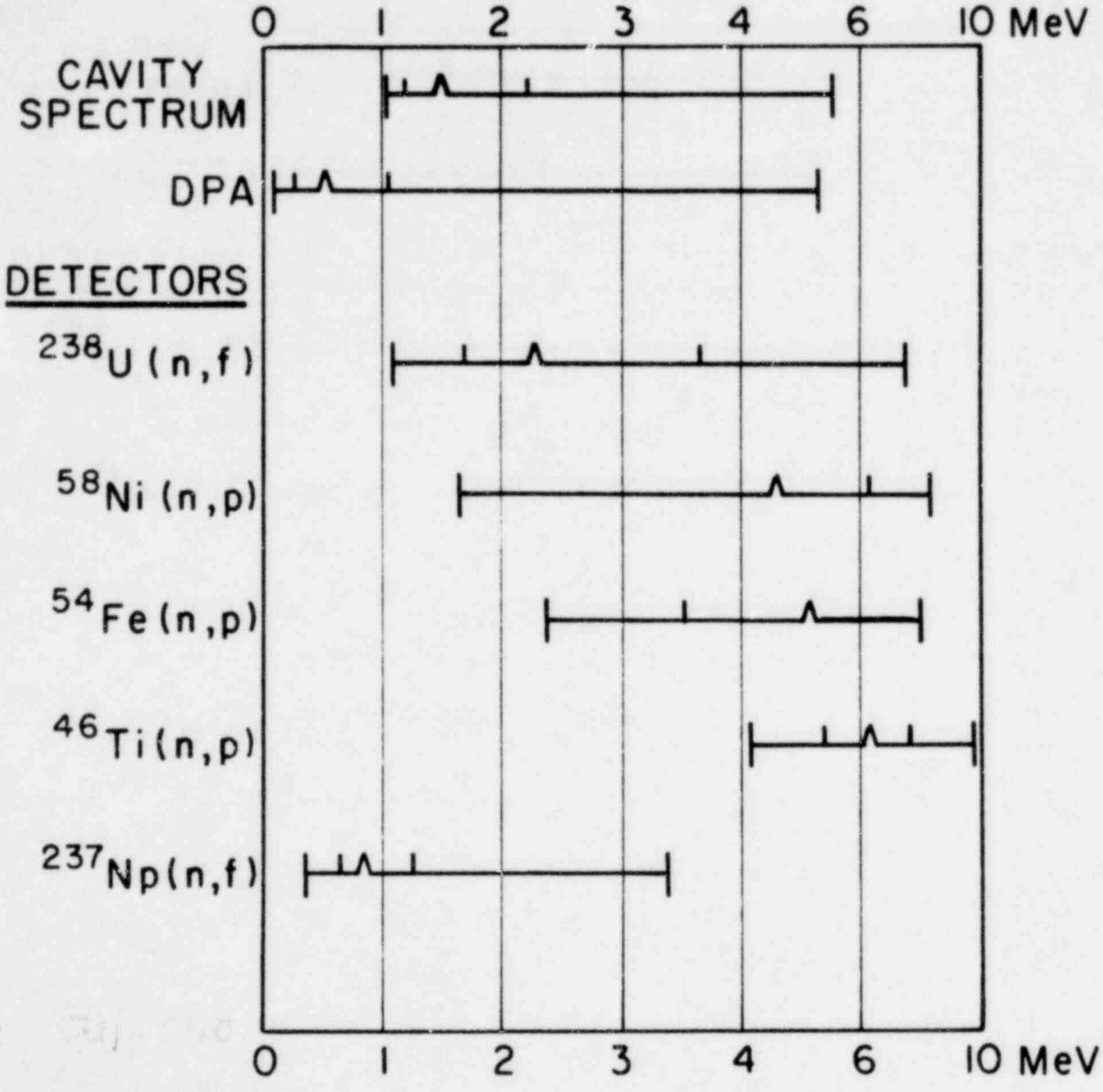

FIGURE 22. Ex-Vessel Results - PWR. Neg 7911456-1

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TABLE 15

ARKANSAS POWER AND LIGHT, UNIT #1 RPV CAVITY FLUENCE (CONVENTIONAL CALCULATION)

REACTION	THRESHOLD	MEASURED REACTION PROBABILITY	FLUENCE ABOVE 1 MeV
^{238}U (n, f)	1.2 MeV	4.45 E-10	1.52 E15
^{58}Ni (n, p)	1.7 MeV	1.30 E-10	1.54 E15
^{54}Fe (n, p)	2.4 MeV	0.91 E-10	1.44 E15
^{46}Ti (n, p)	4.2 MeV	1.78 E-11	1.68 E15
		MEAN VALUE:	1.55 E15
		STD. DEV.:	±6.4% (1σ) ±19% (3σ)

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TABLE 16

ARKANSAS POWER AND LIGHT, UNIT #1 RPV CAVITY FLUENCE (BENCHMARK REFERENCING)

REACTION	THRESHOLD	SPECTRUM COVERAGE FACTOR	FLUENCE ABOVE 1 MeV
^{238}U (n,f)	1.5 MeV	70%	1.47 E15
^{58}Ni (n,p)	2.1 MeV	39%	1.45 E15
^{54}Fe (n,p)	2.5 MeV	22%	1.39 E15
^{46}Ti (n,p)	3.9 MeV	5.7%	1.43 E15
		MEAN VALUE:	1.44 E15
		STD. DEV.:	±2.4% (1σ) ±7.2% (3σ)

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TABLE 17

FULL-POWER VESSEL CAVITY FLUX FOR ARKANSAS POWER AND LIGHT, UNIT #1

	FLUX ABOVE 1 MeV	
FINAL VALUE FROM BENCHMARK REFERENCING PROCEDURES (FOUR DETECTOR AVERAGE WEIGHTED BY SPECTRUM COVERAGE FACTOR)	3.45 E8	1.000
	±3.3% (3σ)	
COMPARABLE VALUES		
CONVENTIONAL CALCULATION, NO BENCHMARK REFERENCING	3.71 E8	1.08
	±19% (3σ)	
BENCHMARK REFERENCING VALUES WEIGHTED BY DPA COVERAGE	3.43 E8	0.99
	±3.3% (3σ)	

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