NUREG/CR-1241, Vol. 4
HEDL-TME 80-6

LWR PRESSURE VESSEL IRRADIATION SURVEILLANCE DOSIMETRY

QUARTERLY PROGRESS REPORT
OCTOBER 1980 - DECEMBER 1980

Hanford Engineering Development Laboratory

Preparation coordinated by
G.L. Guthrie
W.N. McElroy

Prepared for the U.S. Nuclear Regulatory Commission

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NUREG/CR-1241, Vol. 4
HEDL-TME 80-6
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Hanford Engineering Development Laboratory

Operated by Westinghouse Hanford Company
P.O. Box 1970 Richland, WA 99352
A Subsidiary of Westinghouse Electric Corporation

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Manuscript completed: August 1981
Date Published: November 1981

Prepared for Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN No. B5988-7

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SERIES NOTE

Information for the Third Quarter of 1980, (July - September, 1980) is contained in NUREG/CR-1747 (HEDL-TME 80-73), LWR Pressure Vessel Surveillance Dosimetry Improvement Program, 1980 Annual Report, (October 1, 1979 - September 30, 1980), which should be considered as Volume 3 in the NUREG/CR-1241 series.

FOREWORD

The Light Water Reactor Pressure Vessel (LWR-PV) Surveillance Dosimetry Improvement Program has been established by NRC in recognition of the importance of improving, maintaining, and standardizing neutron dosimetry, damage correlation, and the associated reactor analysis data and 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 USNRC supported activities at HEDL, ORNL, NBS, and NRL 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 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. Several subcommittees of ASTM E10 are responsible for the preparation of LWR-PV surveillance standards.

The primary objective of the multilaboratory program is to prepare an updated and improved set of dosimetry, damage correlation, and associated reactor analysis ASTM Standards for LWR-PV 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," 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 the ASTM 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 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 pressure vessel 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-PV irradiation surveillance procedures: (1) proper application of calculational estimates of the neutron fluence delivered to in-vessel surveillance positions, various locations in the vessel wall, and ex-vessel support structures and surveillance positions, and (2) understanding the relationship between material property changes in reactor vessels, in-vessel support structures, and in metallurgical test specimens in test reactors and at accelerated neutron flux positions in operating power reactors.

The first component requires validation and calibration experiments in a variety of neutron irradiation test facilities including LWR-PV mock-ups, power reactor surveillance positions, and related benchmark neutron fields. The benchmarks serve as a permanent measurement reference 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 test reactor "test regions" and operating power reactor "surveillance positions" to locations inside the body of the pressure vessel wall and 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 test 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. Proposed detectors for the program include radiometric detectors, helium accumulation fluence monitors, solid state track recorders, and damage monitors.

The necessity for pressure vessel mock-up facilities for 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 mock-up are in progress. The low flux version is known as the Poolside Critical Assembly (PCA) and the high flux version is known as the Pool Side Facility (PSF). Both are located at ORNL. As specialized benchmarks, these facilities will provide well-characterized neutron environments where active and passive neutron dosimetry, various types of LWR-PV neutron field calculations, and temperature-controlled metallurgical damage exposures are brought together.

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 calls for adherence to several ASTM Standards which require establishment of a surveillance program for each power reactor and incorporation of flux monitors and post-irradiation neutron field evaluation. Revised and new standards in preparation will be carefully structured to be up-to-date, flexible, and, above all, consistent.

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FIGURES

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ACKNOWLEDGMENTS

The following organizations presently participate in the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program and will periodically contribute written reports, experimental data, or calculations.

Atomic Energy Research Establishment, Harwell, England (AERE-H)

Babcock & Wilcox Company (B&W), USA

Battelle Memorial Institute (BMI), Columbus Laboratory, USA

Brookhaven National Laboratory (BNL), USA

Centre d'Etude de l'Energie Nucleaire - Studiecentrum Voor Kernenergie (CEN/SCK), Mol, Belgium

Centre d'Etudes Nucleaires de Saclay (CEA, Saclay), Gif-sur-Yvette, France

Combustion Engineering, Inc. (CE), USA

EG&G ORTEC, USA

Electric Power Research Institute (EPRI), USA

Fracture Control Corporation (FCC), USA

General Electric Vallecitos Nuclear Center (GE-VNC), USA

Hanford Engineering Development Laboratory (HEDL), USA

Institut für Kernenergetik und Energiesysteme der Universität Stuttgart (IDE), Stuttgart, Germany

IRT Corporation (IRT), USA

Italian Atomic Power Authority (ENEL), Italy

Japan Atomic Energy Research Institute (JAERI), Japan

Kernforschungsanlage Jülich Gmbh, Germany (KFA)

Kraftwerk Urion, Germany

Macalester College (MC)

National Bureau of Standards (NBS), USA

Naval Research Laboratory (NRL), USA

Oak Ridge National Laboratory (ORNL)

Radiation Research Associates (RRA), USA

Rockwell International Energy Systems Group (RIES)

Rolls-Royce and Associates Limited, Derby, England (RRAL)

Science Applications Incorporated (SAI), USA

Ship Research Institute (SRI), Japan

Southwest Research Institute (SWRI), USA

University of Arkansas (UA), USA

University of California, Santa Barbara (UCSB), USA

University of Tennessee (UT), USA

University of Tokyo, Japan

Westinghouse Electric Corporation - Nuclear Technology Division (WEC-PWR-SD)

Westinghouse Electric Corporation - Research and Development Division (WEC-RD)

SUMMARY

HANFORD ENGINEERING DEVELOPMENT LABORATORY (HEDL)

The FERRET computer code has been used to assign uncertainties to the re-evaluated calculations of the flux (Ø, E > 1 MeV) and displacement rate (dpa/sec) for 19 surveillance capsules. The work differs from that previously reported in that (1) the FERRET code is used rather than a revised SAND-II code, and (2) the uncertainties are reported, as derived from the input covariance information. It is found that the uncertainty in the flux is 11-21% (1σ) and the uncertainty in the displacement rate is 9-16% (1σ). Results agree reasonably well with those in the previous report.

A statistical study has been conducted to determine if the shape of the Charpy embrittlement trend curve depends on the copper or nickel content of the specimens under irradiation (chemistry dependent saturation effect). A formula of the type:

$$\Delta NDTT(Charpy) = f(Chemistry) \times (Øt)^N$$

was used in a least squares fitting routine, with the fluence exponent being linearly dependent on the copper or nickel concentration. Several sets of data were analyzed. The F statistic was used to determine the significance of the reductions in the standard deviations. Data from the British Herald reactor showed a possible relationship between the value of the fluence exponent N and the copper content of the specimen under irradiation.

Emulsion exposures conducted in the LWR-PV at the PCA at the Oak Ridge National Laboratory are presented. A new method of scanning emulsions in the integral mode is advanced. The integral mode response of nuclear research emulsions is analyzed and sources of uncertainty are established.

OAK RIDGE NATIONAL LABORATORY (ORNL)

Program administration activities relative to promulgation of status and planning for future meetings were accomplished. Several papers regarding status were presented at the Eighth Water Reactor Safety Research Information Meeting in Gaithersburg, Maryland, Oct. 27-31, 1980.[1-3] Coordination and planning activities for the Fourth ASTM-EURATOM International Symposium on Reactor Dosimetry and the IAEA Specialists' Meeting, CAPRICE 82, were conducted at Lynchburg, Virginia, Dec. 9-10, 1980 and at ORNL.

The measurements and analysis of the dosimetry experiment for the BSR-HSST-IV irradiation series were completed and documented for capsules A and B.[2] Documentation and data compilation for the NRC Computational "Blind Test" were completed.[3] A question regarding the significance of transverse leakage corrections was resolved based on Monte Carlo calculations.[4] Leakage treatments by the participants were adequate. Documentation and analyses of ORR-PSF irradiation capsule data were also continued. Sketches (cf. Figure 1) for future metallurgical data analysis were prepared and data from the pressure vessel capsule were processed.

A critical review of the Standard Practice for the Analysis of Charpy Impact Data was completed and a new procedure was developed. The "ASTM Guide on the Application of Neutron Transport Methods" was reviewed by ORNL personnel and submitted to appropriate ASTM committee members for further review.

CEN/SCK (BELGIUM)

Several reports from the CEN/SCK laboratories at Mol, Belgium are attached as an appendix.

Hanford Engineering Development Laboratory (HEDL)

A. RE-EVALUATION OF DOSIMETRY FOR 19 PWR SURVEILLANCE CAPSULES - II

R. L. Simons
Hanford Engineering Development Laboratory

OBJECTIVE

The primary objective of this program is to improve the dosimetry analysis and procedures for correlating data from surveillance of light water reactor (LWR) pressure vessels. These improvements will be incorporated in the writing of ASTM standards for the correlation of irradiation induced damage to LWR pressure vessel steels. Before these standards can be adopted they must be tested with actual surveillance data. The purpose of this report is to present the latest results of an effort to improve the analysis of the dosimetry from 19 Westinghouse pressurized water reactor (PWR) surveillance capsules.

SUMMARY

Using the FERRET computer code a preliminary uncertainty analysis of re-evaluated fluxes for PWR surveillance capsules shows that the flux ($E > 1$ MeV) can be determined to within 11-21% (1σ) and the displacement rate can be determined to within 9-16% (1σ). The lower uncertainty results from using accurately determined fission rates for Np and U. The fluxes and displacement rates determined with the FERRET code are in reasonable agreement with values determined previously using the modified SAND II code.

ACCOMPLISHMENTS AND STATUS

A previous report[1] gave results of fluence ($E > 1$ MeV) and dpa from re-evaluation of the dosimetry from 19 Westinghouse pressurized water reactor (PWR) surveillance capsules. It was found that, on the average, the neutron exposures were 40% higher than reported in the original surveillance reports. This was in part due to differences in cross sections and neutron

spectra used to evaluate the dosimetry. An important aspect of reporting neutron exposures is to state the corresponding uncertainty in the exposures. This report describes initial efforts to determine uncertainties in the previously reported neutron exposures. This uncertainty analysis is limited to capsules A and D of San Onofre Unit 1.

The modified SAND II code incorporating the use of a priori neutron spectrum information was used in the previously reported re-evaluation[1] of the surveillance dosimetry. The FERRET code[2] (a generalized least squares approach), which also includes a priori spectrum information, was used in the present analysis to determine the uncertainty in the fluxes and displacement rates. The FERRET code has the advantage of properly handling correlations in the cross sections and spectra. (The MONTE CARLO version of SAND II can also be developed to handle correlation terms properly.[3] However, FERRET is less costly to run than MONTE CARLO, primarily due to the reduced number of energy groups.)

There are three primary quantities that affect the uncertainty in the solution spectrum and integral fluxes: uncertainties in the reaction rates, cross sections, and the a priori neutron spectrum. The uncertainty analysis in the present report is a preliminary one intended to reveal trends. The cross sections and their uncertainties and correlations were not normalized to a standard neutron field such as the fission spectrum. The uncertainty in the a priori neutron spectrum was nominally set at $\pm$30% for all neutron groups with correlations extending over five neutron groups assuming a gaussial distribution. More rigorous definition of the correlations for the a priori spectrum will be included after completion of the analysis of the spectra measured in the pressure vessel mock-up in the pool critical assembly (PCA). The reaction rates and their uncertainties are listed in Table HEDL-1. The uncertainties were determined from the standard deviation of multiple measurements and the uncertainty in the gradient correction, or they were taken as $\pm$10% for single measurements. Exceptions were the two fission foils in capsule A. The fission reaction rates were found to be inconsistent with the rest of the reaction rates and the a priori spectrum by 25-50%; consequently, the uncertainties were assigned high values.

TABLE HEDL-1

INTEGRAL REACTION RATES AND UNCERTAINTIES FOR SAN ONOFRE
UNIT 1 SURVEILLANCE CAPSULES A AND D

	CAPSULE A		CAPSULE D	
Reaction	Rate(dps/nucleus)	% Uncertain	Rate(dps/nucleus)	% Uncertain
Fe54(n,p)	2.03×10^{-14}	5	2.58×10^{-14}	8
Co59(n,γ)	1.64×10^{-11}	10	2.04×10^{-11}	3
Co59(n,γ)[Cd]	6.16×10^{-12}	6	7.81×10^{-12}	6
Ni58(n,p)	2.46×10^{-14}	10	-	-
Np237(n,f)[Cd]	5.43×10^{-13}	50	-	-
U238(n,f)[Cd]	9.09×10^{-14}	25	-	-
Cu63(n,α)	1.50×10^{-16}	8	1.34×10^{-16}	11

Table HEDL-2 shows results of the FERRET data analysis. The fluxes for capsule A are about three-fourths those of capsule D. This was noted in the original surveillance report and was attributed to differences in fuel loading and burnup.[4] The uncertainties in the fluence (E > 1 MeV) are greater than the uncertainties in the displacement rate. This is due to the relatively small uncertainty in the neutron spectrum above the $^{54}Fe(n,p)^{54}Mn$ threshold where a relatively large fraction of the displacements occurs but only a small fraction of the integral flux (E > 1 MeV).

The uncertainties in the fast flux and dpa/sec for capsule A are less than that for Capsule D. This is due to the inclusion of the two fission reactions of ^{237}Np and ^{238}U in the calculation for capsule A. A second case was run for capsule A with modified fission rates and with the input uncertainty in the fission rates reduced to 10%. This produces nearly a factor of two reduction in the resulting uncertainty in flux and dpa/s for capsule A relative to capsule D. This demonstrates the importance of including accurate fission reaction rates in the dosimetry for surveillance capsules. It is also important to compare the single dosimeter results (such as for a fission foil) with both calculated and generic reaction rates for comparable surveillance capsules in order to guard against gross errors.

The SAND II results previously reported differ from the present analysis using FERRET by 3% for capsule D and 30% for capsule A. The large difference for capsule A is due in part to a substantial increase in the uncertainty assigned the fission foils in the more recent capsule A analysis.

EXPECTED ACCOMPLISHMENTS IN THE NEXT REPORTING PERIOD

Future work will include completion of the error analysis of the remaining seventeen spectra. Dosimetry analysis of additional PWRs will continue.

TABLE HEDL-2

INTEGRAL FLUXES AND DISPLACEMENT RATES AND THEIR UNCERTAINTIES (1σ) FOR SAN ONOFRE UNIT 1

Surveillance Capsule	Integral Flux (10^{12} /cm^2/sec) E >0 MeV	Integral Flux (10^{12} /cm^2/sec) E >1 MeV	Displacement Rate (10^{-10} dpa/sec)
A	2.9(±17%)	0.35(±17%)	6.1 (±13%)
D	4.0(±12%)	0.54(±21%)	8.8 (±16%)
A*	2.9(±10%)	0.36(±11%)	6.3 (±9%)

*Fission rates adjusted upward and uncertainty in fission rates reduced to 10%.

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B. AN INVESTIGATION OF THE DEPENDENCE OF CHARPY TREND CURVE SATURATION ON NICKEL OR COPPER CONCENTRATION

George L. Guthrie
Hanford Engineering Development Laboratory

Objective

The objective of this work is to determine whether the metallurgical composition of Charpy surveillance specimens has an effect on the value of the fluence exponent in embrittlement predictive formulas of the type:

$$\Delta T \text{ (Charpy shift)} = \text{(linear function of chemistry)} \times \left(\frac{\phi t}{10^{19}}\right)^{N}. \qquad (1)$$

We wish to ascertain whether variations in surveillance specimen chemistry affect the shape of the "trend" curve, thus fixing the degree of apparent "saturation". This information is useful in the writing of ASTM standards for (a) Charpy embrittlement trend laws, and (b) damage correlation. The writing of these standards is required as part of the LWR surveillance dosimetry improvement program.

Summary

Charpy-shift-vs-fluence data (30ft.lb. Charpy energy) were used in a statistical least squares parameter adjustment study involving functions of the type:

$$\Delta T \text{ (Charpy shift)} = \text{(linear function of chemistry)} \times \left(\frac{\phi t}{10^{19}}\right)^{N}. \qquad (2)$$

In these studies, N was assumed to be of the form (a) N = Constant, (b) N = A + B • Cu (wt%) or (c) N = A + B • Ni (wt%) where "Constant", A, and B are adjustable parameters.

Several data sets were studied and statistical F tests were used to evaluate the results. It was concluded that the studies did not indicate that the exponent N was dependent on the Cu or Ni concentration for most of the data sets investigated. A possible exception was the case of a copper dependence for N for a data set involving irradiations in the British Herald reactor. For this particular subset, it was found that the inclusion of a copper dependence for N improved the fit of the data, but the noted improvement could occur 8% of the time even in the absence of a real physical effect.

Accomplishments and Status

One of the more widely used trend-curve laws for irradiation induced embrittlement (Charpy shift) is a law of the general type:

$$\Delta T \text{ (Charpy shift)} = \text{(function of specimen chemistry)} \times \left(\frac{\phi t}{10^{19}}\right)^{N} \qquad (3)$$

Previous work[1] has shown that the type of formulation shown in Eq (3) fits the available data as well as more complicated functional forms which contain explicit features representing coarsening or annealing effects. In the simple equation of the type shown in Eq (3), the exponent is expected to satisfy the inequality $0 < N < 1$. Laws containing low values of N exhibit a more prominent knee in the trend curve, a feature which may be associated with the presence of a saturation effect. There is currently speculation as to whether saturation is more apparent in alloys containing high levels of nickel and/or copper.

This would indicate that it would be worthwhile to look for a copper (or Ni) dependence for the fluence exponent "N" in least squares fits of Charpy shift data. Choosing the simplest possible copper dependence for the exponent, we assume:

$$\Delta T = \text{(function of chemistry)} \times \left(\frac{\phi t}{10^{19}}\right)^{(A + B \cdot Cu)} \qquad (4)$$

A similar formula was also investigated for a determination of a Ni saturation effect. This is shown as Eq (5).

$$\Delta T = (\text{function of chemistry}) \times \left(\frac{\phi t}{10^{19}}\right)^{(A + B \cdot Ni)} \qquad (5)$$

The data collections investigated were as follows:

(1) The weld data (Charpy shift vs fluence and chemistry) used by Varsik and Byrne.[2] This table was augmented by the addition of several Charpy specimen weld data points from surveillance capsules in Westinghouse reactors, using the adjusted fluence values found by R. L. Simons.[3]

(2) The Herald reactor data used by T. J. Williams and R. L. Squires.[4]

(3) The Dido-Pluto data base used by T. J. Williams and R. L. Squires.[4]

(4) A combination of (2) and (3) above.

(5) The separate weld data from the Westinghouse reactors (part of (1), above).

The British[4] and U. S. data[2,3] could not be combined because the two sets were accumulated in different temperature regions. The British irradiations were all at 482°F while the U. S. irradiations were principally at or near 550°F. The limited data collection in Set 5 above was treated separately in the hope that it might be more self consistent and reveal physical laws hidden in the scatter of the large sets.

The chemistry-dependent pre-multiplier part of the Charpy shift formula [Eq (3)] was chosen to be a linear function of the type:

$$f = \sum_i A_i El_i + \left(\sum_j b_j El_j\right) \cdot Cu \text{ (wt\%)} \qquad (6)$$

where the A_i and b_j are adjustable constants to be determined by a least squares procedure. El_i is the wt% of the i th element in the alloy (and similarly for j). The sum on i (but not j) is meant to include a constant "unity" element, to provide an additive but adjustable constant.

Several different formulas of the type of Eq (4) were used, with only a limited number of terms being used in each formula. The choice of which terms to include in each sum and which to leave out is made basically by observing which terms in the sums have values that correlate with the embrittlement sensitivity $\Delta T/(\phi t)^N$. However, in detail, there is no accepted best way to choose a proper set.[5]

One such set used in the present study was Ni, Cu, Cu^2 and an additive constant. This set was chosen by the IMSL routine RLSEP[6] which is a "forward stepwise linear regression algorithm" similar to that described in chapter 6 of Draper and Smith.[5]

Another set of summation terms chosen was P, Ni, Cr, Mo, Cu, Cu•C, Cu•Mn, Cu•P, Cu•S, Cu•Si, Cu•Ni, Cu•Mo, Cu^2 and an additive constant. This set was chosen by observing the correlations of the individual terms with $\Delta T/(\phi t)^{.255}$ in the combined data bank consisting of the Varsik-Byrne[2] weld data and the Westinghouse surveillance data.[3] All elemental terms with a correlation $\geq .2$ and all (Cu•Element) product terms with a correlation $\geq .25$ were included in the analysis. The results of the least-squares fits are shown in Table HEDL-3, and an analysis of the results is summarized in Table HEDL-4. Use of fluence exponent values slightly different than .225 in the correlation study did not affect the choice of terms to be included in the least squares fits.

TABLE HEDL-3

LEAST-SQUARES FITS OF THE CHARPY SHIFT USING A CHEMISTRY-DEPENDENT FLUENCE EXPONENT

COMPUTER RUN NUMBER	DATA POINTS	DEGREES OF FREEDOM*	STANDARD DEVIATION	EXPONENT LINEARLY DEPENDENT ON:	CHEMISTRY DEPENDENCE OF PREMULTIPLIER
1	15 HERALD	10	12.4°C	Adjustable constant	Ni, Cu, Cu^2, and additive constant
2	15 HERALD	9	7.715°C	Copper and additive constant	Ni, Cu, Cu^2, and additive constant
3	15 HERALD	9	10.12°C	Nickel and additive constant	Ni, Cu, Cu^2, and additive constant
4	25 DIDO-PLUTO	20	21.41°C	Adjustable constant	Ni, Cu, Cu^2, and additive constant
5	25 DIDO-PLUTO	19	21.49°C	Copper and additive constant	Ni, Cu, Cu^2, and additive constant
6	25 DIDO-PLUTO	19	21.41°C	Nickel and additive constant	Ni, Cu, Cu^2, and additive constant
7	Varsik and Byrne Welds plus West. Surv. Welds	46	17.74°C	Adjustable constant	P, Ni, C, Mo, Cu, plus product of copper concentration with C, Mn, P, S, Si, Ni, Mo, and Cu. Also an additive constant

*Number of data points minus number of adjustable parameters.

TABLE HEDL-3 (Cont'd)

COMPUTER RUN NUMBER	DATA POINTS	DEGREES OF FREEDOM*	STANDARD DEVIATION	EXPONENT LINEARLY DEPENDENT ON:	CHEMISTRY DEPENDENCE OF PREMULTIPLIER
8	Varsik and Byrne Welds plus West. Surv. Welds	45	17.92°C	Nickel and additive constant	Same as run seven
9	Varsik and Byrne Welds plus West. Surv. Welds	45	17.51°C	Copper and additive constant	Same as run seven
10	West. Surv. Welds (RLS)	6	16.79°C	Adjustable constant	Ni, Cu, Cu^2 and additive constant
11	West. Surv. Welds (RLS)	5	18.31°C	Nickel and additive constant	Ni, Cu, Cu^2 and additive constant
12	West. Surv. Welds (RLS)	5	16.94°C	Copper and additive constant	Ni, Cu, Cu^2 and additive constant

TABLE HEDL-4

ANALYSIS OF THE F STATISTIC APPLIED TO LEAST-SQUARES FITS

RUNS COMPARED FROM TABLE I	LINEAR PREMULTIPLIER	LOOKING FOR SIGNIFICANCE OF NON-ZERO "B" FORMULA	LOOKING FOR FLUENCE EXPONENT DEPENDENCE ON	F VALUE	SIGNIFICANCE OF F VALUE
1, 2	Linear combination of Ni, Cu, Cu^2 and an additive constant	$N = A + B(Cu-Cu_{AVE})$	Cu	$\left(\frac{12.4}{7.72}\right)^2 = 2.58$	Improvement could happen approximately 8% of time by chance
1, 3	SAME AS ABOVE	$N = A + B(Ni-Ni_{AVE})$	Ni	$\left(\frac{12.4}{10.12}\right)^2 = 1.50$	Improvement could happen > 25% of time by chance
4, 5	SAME AS ABOVE	$N = A + B(Cu-Cu_{AVE})$	Cu	$\left(\frac{21.41}{21.49}\right)^2 = .993$	"Improvement" could happen > 50% of time by chance
4, 6	SAME AS ABOVE	$N = A + B(Ni-Ni_{AVE})$	Ni	$\left(\frac{21.41}{21.41}\right)^2 = 1.0$	"Improvement" could happen ~ 50% of time by chance
7, 8	Linear combination of P, Ni, Cr, Mo, Cu, Cu·C, Cu·Mn, Cu·, Cu·S, Cu·Si, Cu·Ni, Cu·Mo, Cu^2 and a constant	$N = A + B(Ni-Ni_{AVE})$	Ni	$\left(\frac{17.74}{17.92}\right)^2 = .980$	"Improvement" could happen > 50% of time by chance
7, 9	SAME AS ABOVE	$N = A + B(Cu-Cu_{AVE})$	Cu	$\left(\frac{17.74}{17.51}\right)^2 = 1.03$	Improvement could happen > 45% of time by chance
10, 12	Linear combination of Ni, Cu, Cu^2, and an additive constant	$N = A + B(Cu-Cu_{AVE})$	Cu	$\left(\frac{16.79}{16.94}\right)^2 = 0.98$	"Improvement" could happen > 50% of time by chance
10, 11	SAME AS ABOVE	$N = A + B(Ni-Ni_{AVE})$	Ni	$\left(\frac{16.79}{18.31}\right)^2 = .84$	"Improvement" could happen > 50% of time by chance

The strongest indication of a chemistry effect on N (fluence exponent) was found for the British Herald reactor data set.[4] For this set the F test showed that the apparent improvement in the standard deviation could occur by chance 8% of the time even in the absence of any physical effect. We therefore conclude that there may be an effect of chemical composition on the shape of the trend curve, but the data sets investigated did not show any strong evidence of its existence.

Expected Future Accomplishments

Similar studies may be conducted as additional data becomes available, but the schedule is uncertain.

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C. LWR-PV NEUTRON SPECTROMETRY WITH NUCLEAR RESEARCH EMULSIONS

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Objective

Neutron spectrometry plays a crucial role in understanding the limitations of light water reactor pressure vessels (LWR-PV). Within the ASTM Master Matrix for LWR Pressure Vessel Surveillance Standards, neutron spectrometry is essential in the following standards:

I-A: "Analysis and Interpretation of Nuclear Reactor Surveillance Results"

I-B: "Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials"

I-C: "Surveillance Test Results Extrapolation"

II-A: "Application of Neutron Spectrum Adjustment Methods"

II-E: "Benchmark Testing of Rea vessel Dosimetry"

The energy dependence of damage produced by neutrons in LWR-PV steel has been recognized for some time now.[1-3] However, differential measurement of neutron energy spectra in LWR-PV environments has not been possible heretofore. Inherent limitations prevent differential techniques from being applied in power reactor environments.[4] Integral measurements which can be conducted in power reactors, such as radiometric, SSTR, or HAFM experiments, possess very limited energy resolution. Of even greater significance is that neutron induced radiation damage in iron is non-negligible in the 0.1-1.0 MeV energy region for these LWR-PV environments. Unfortunately, these passive high power techniques offer little coverage in this energy

region. Hence the establishment of a low power LWR-PVS benchmark at the PCA has offered a unique opportunity for progress in the LWR-Pressure Vessel Surveillance Dosimetry Program. The overall advantages of benchmark field referencing has been expounded in a recent symposium.[5]

In operating LWR, neutron induced damage in the PV arises predominantly in the energy range 0.01-6.0 MeV. Fortunately, a number of differential neutron spectrometry techniques already exist which can adequately cover this energy region when applied in consort.[4] In particular, gas filled proton-recoil proportional counter spectroscopy, proton-recoil nuclear research emulsions, and $^{6}Li(n,\alpha)$ fast neutron spectroscopy can be used to cover this energy region in the PCA. This report covers efforts with proton-recoil nuclear research emulsions.

Summary

Emulsion exposures conducted in the LWR-PVS at the PCA at Oak Ridge National Laboratory are presented. A new method of scanning emulsions in the integral mode is advanced. The integral mode response of nuclear research emulsions is analyzed and sources of uncertainty are estimated.

Accomplishments and Status

Introduction -- In LWR-PV environments, neutron-induced radiation damage arises predominantly in the 0.01-6.0 MeV energy region. No single method of in-situ neutron spectrometry exists which can completely cover this energy region.[4] Work with proton-recoil proportional counters has not been extended beyond a few MeV, due to the escape of more energetic protons from the finite sensitive volume of the counter. In fact, corrections for such finite size effects of in-situ proportional counters can be non-negligible above 0.5 MeV.[6] In the LWR-PVS at the PCA, proton recoil proportional counter spectrometry efforts have revealed that very significant finite-size corrections are mandatory.[7]

In this regard, it is fortunate that nuclear research emulsions afford a method of fast neutron spectrometry that complements proportional counter spectrometry. Finite-size effects are much more manageable in emulsions because of the reduced range of recoil protons. As a consequence, nuclear emulsion fast neutron spectrometry has been applied at energies up to 15 MeV.[8] For in-situ spectrometry in LWR environments, emulsion measurements up to 6.0 MeV are possible with very small finite-size corrections. Since the hydrogen concentration of nuclear research emulsions is well known, absolute measurements are possible.

It is possible to use emulsion data to obtain both differential and integral spectral information. Emulsion work is customarily carried out in the differential mode.[9,10] In contrast, emulsion work in the integral mode is a new concept and, therefore, a fuller explanation of this approach is included below. In this integral mode, emulsions provide absolute integral reaction rates that can be used in spectral adjustment codes.[11] In the past, such adjustment codes have not utilized integral reaction rates based on emulsions. The beauty of emulsion integral reaction rates is their tie to the elastic scattering cross section of hydrogen. This $\sigma_{n,p}(E)$ cross section is universally accepted as a standard cross section and is known to an accuracy of roughly 1%. Hence, emulsion integral reaction rates will afford a significant new dimension for work with spectral adjustment codes.

Emulsion irradiations in the LWR-PV are described in the section immediately below. In the follow-on sections, the method of nuclear emulsion spectrometry is presented with special emphasis given to new track scanning and data analysis techniques.

Emulsion Irradiations in the LWR-PV

Emulsion irradiations in the 8/7 and 12/13 configurations were carried out in August 1979. Table HEDL-5 summarizes these emulsion exposures.

TABLE HEDL-5

NUCLEAR EMULSION EXPOSURES IN THE 8/7 AND 12/13 CONFIGURATIONS

Irradiation Number	LWR-PV Configuration	Location*	Power** Level (W)	Duration*** (min)	Emulsion ID
1	8/7	1/4 T	5	20	PCA-1A
2	8/7	1/2 T	12	15	PCA-2A
3	8/7	3/4 T	25	15	PCA-3A
4	12/13	1/4 T	35	20	PCA-4A
5	12/13	1/2 T	85	15	PCA-5A
6	12/13	3/4 T	175	15	PCA-6A

*All exposures at reactor midplane.
**Power determined from PCA reactor instrumentation that possessed a combined linearity and reproducibility of approximately 3%.
***Exposure time uncertainty = ±5 s for all irradiations.

For the 4/12 configuration, emulsion irradiations were carried out in October 1980. In all 4/12 configuration exposures, as summarized in Table HEDL-6, the SSC was in position and the BSR was shut down.

Each emulsion package contained an ~200-μm and an ~400-μm Ilford L-4 emulsion. These emulsions were wrapped with two layers of paper to prevent exposure to light. This wrapping also provided for a snug fit inside the dummy proton-recoil chambers so that the emulsions could be oriented with confidence. Figure HEDL-1 displays the in-situ configuration used for these emulsions irradiations.

TABLE HEDL-6

NUCLEAR EMULSION EXPOSURES IN THE 4/12 SSC CONFIGURATION

Irradiation Number	Location	Power Level (W)	Time In	Duration (min)	Integrated Power (W-min)	Emulsion Label
1	1/4 T	2.00	1537 (10/27)	12.50	25.00	B1
2	1/4 T	8.00	1600 (10/27)	12.50	100.0	B3
3	1/2 T	4.00	0840 (10/28)	12.50	50.0	B4
4	1/4 T	2.00	0910 (10/28)	12.50	25.00	B6
5	1/4 T	4.00	0938 (10/28)	25.00	100.0	B7
6	1/2 T	4.00	1012 (10/28)	12.50	50.0	B5
7	1/2 T	8.00	1031 (10/28)	25.00	200.0	B8
8	3/4 T	8.00	1104 (10/28)	12.50	100.0	B9
9	3/4 T	16.00	1122 (10/28)	25.00	400.0	B10
10	VB	20.00	1211 (10/28)	44.00	880.0	B11
11	VB	20.00	1303 (10/28)	11.00	220.0	B12
12	1/4 T off	2.00	1322 (10/28)	15.00	30.00	B13

Experimental Notes for Table HEDL-6:

All irradiations were at midplane. The dummy proton-recoil chamber was 2-mm longer than the proton-recoil chamber, so that the emulsion center was 1-mm higher than the PCA midplane for the 1/8 T, 1/2 T, and 3/4 T locations. All dummy proton-recoil chambers were mounted as identically as possible to the proportional counter neutron spectrometry measurements, e.g., the preamp was in place during all runs. A modified proton-recoil chamber was used in the VB. All V6 emulsions are centered around midplane. Runs 1-3 were end on towards the PCA face. Runs 4-12 had emulsions oriented parallel to the PCA face. Emulsion labels were on the side towards the core with the arrows pointing down.

Irradiation 1 -- Emulsion was 6-in. high for 3 seconds (included in 12,50 minutes). The alignment was completed in 20 seconds.

Irradiation 2 -- Dropped into alignment after 17 seconds.

Irradiation 3 -- Dropped into alignment after 60 seconds.

Irradiation 4 -- Dropped into alignment after 10 seconds.

Irradiation 5 -- Dropped into alignment after 1.5 seconds.

Irradiation 6 -- Dropped into alignment after 6 seconds.

Irradiation 7 -- Dropped into alignment after 7 seconds.

Irradiation 8 -- Dropped into alignment after 3 seconds.

Irradiation 9 -- Dropped into alignment after 2 seconds.

Irradiations 10 & 11 -- Used the proportional counter VB mount. No alignment slot. Direct visual alignment.

Irradiation 12 -- Dropped into alignment after 10 seconds.

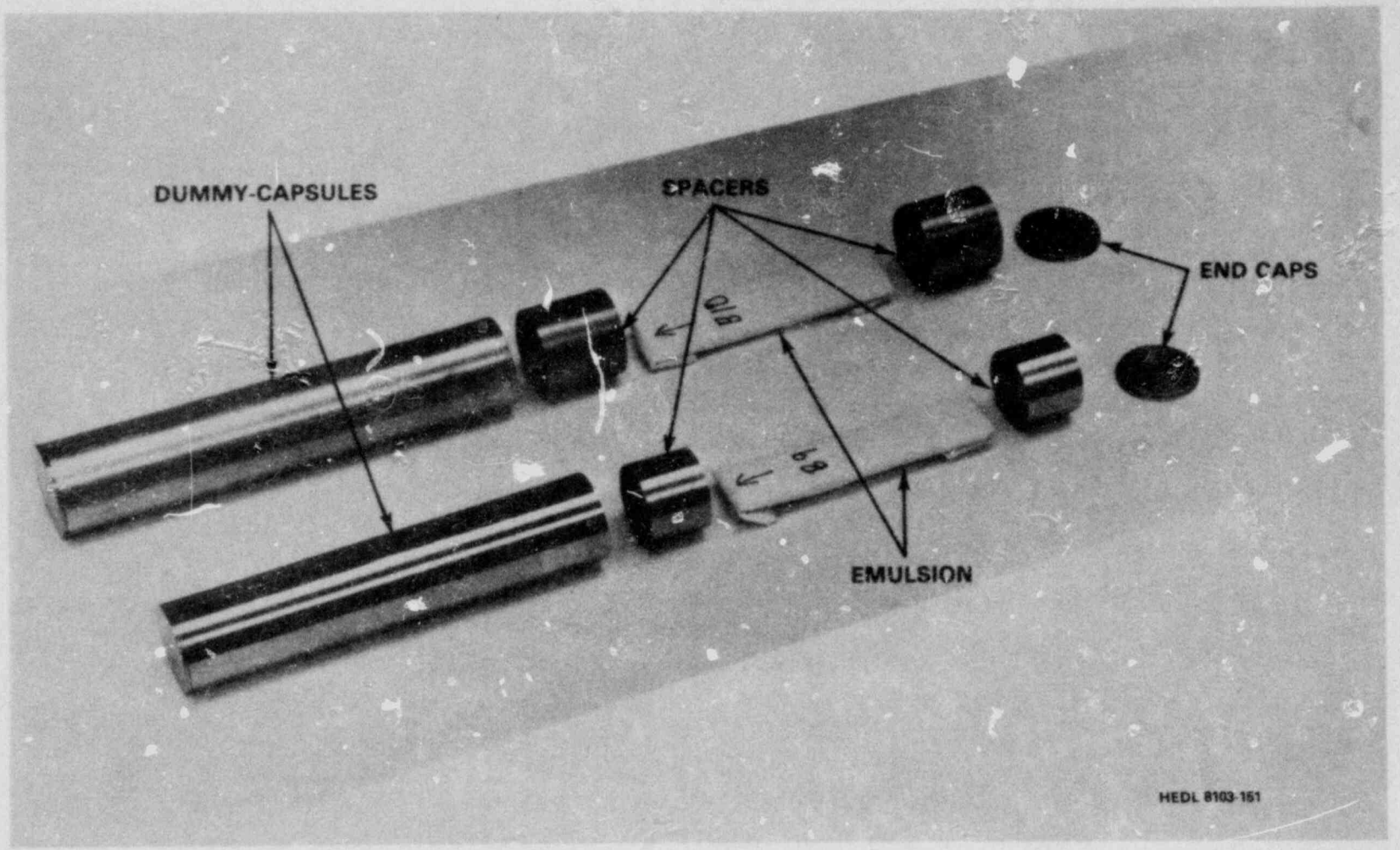

FIGURE HEDL-1. Dummy Proportional Counter Capsule Assemblies Used for In-Situ Emulsion Irradiations in LWR-PVS. Neg P10254-3

Care was exercised in defining the in-situ emulsion orientation for all of these irradiations. In contrast with active spectrometry techniques, emulsions afford information on the angular neutron flux.[12] Hence, knowledge of the in-situ orientation of the emulsion is warranted. To this end, all emulsion packages were labeled with an emulsion package number and an arrow to permit loading with a known in-situ orientation.

For irradiations 1-3 of the 4/12 configuration emulsion packages were placed end on toward the core with the arrow pointing downward. In all other irradiations, emulsion packages were oriented parallel to the PCA core face so that the label faced toward the PCA core and the arrow pointed downward.

Fast neutron spectrometry with nuclear research emulsions is a well established method.[9,10] Hence it is only necessary to describe recent advances in this method. To this end, operation in the differential mode is briefly reviewed. In addition, the new concept of proton-recoil emulsion observations in the integral mode is explained in some detail. Finally, uncertainty estimates are presented for integral mode emulsion neutron dosimetry.

Operation in the Differential Mode -- As stressed in the Introduction, the most accurately established nuclear cross section for neutron measurements is the neutron-proton scattering cross section $\sigma_{n,p}(E)$. Nuclear Research Emulsion (NRE) manufactured by Ilford Limited contains a controlled quantity of hydrogen per unit volume (0.053 ± 0.001 g/cm^3 at a relative humidity of 55%). After irradiation in a fast neutron environment, the proton tracks are revealed with suitable emulsion processing. When proton-recoil track ranges (lengths) are measured, the range distribution can be transformed into a proton-recoil energy spectrum by making use of the well-known proton range-energy relationship for the emulsions.[13] It has been shown[9] that a neutron fluence-spectrum $[\overline{\phi(E)t}]$ can be deduced from the equation:

$$[\overline{\phi(E)}t] = \frac{-E}{\sigma_{np}(E)n_p} \frac{dM(E)}{dE} \quad , \qquad (1)$$

where $\overline{\phi(E)}$ is the average neutron flux in neutrons/cm^2 per MeV per sec, t is the exposure time, E is the neutron or proton energy in MeV, n_p is the atomic hydrogen density in the emulsion in atoms/cm^3, and M(E) is the proton spectrum in proton tracks per MeV per cm^3 observed in the emulsion at energy E. Note that $[\overline{\phi(E)}t]$ depends upon the slope of the proton spectrum at energy E. This means that observation of roughly 10^4 tracks will yield statistical accuracy of the order of 10% in the neutron spectrum fluence along with energy resolutions also of the order of 10%. Spectral measurements determined with NRE are absolute. Details of the method are well documented in the literature.[9,10,14]

Since it is often not possible to determine the direction of the short proton tracks, tracks with both ends in the volume scanned must be treated differently from those with one end outside the volume scanned. If both ends are in the volume scanned, it is certain that the n-p scattering occurred in that volume, and the tracks are assigned a statistical weight of one. If one end is outside the volume scanned, the track is assigned a statistical weight of 0.5, since the probability that it started in that volume is just 0.5.

Figure HEDL-2 shows a weighted proton track length distribution per cm^3 measured in emulsions irradiated at the center of a reactor core. Through the range-energy relationship, this distribution can be transformed into a proton energy spectrum as illustrated in Figure HEDL-3. This proton energy spectrum is fit piecewise to first- or second-degree polynomials in order to obtain a "first-guess" of the neutron fluence spectrum from Eq. (1).

This "first-guess" neutron spectrum can be further refined using the integral relation:

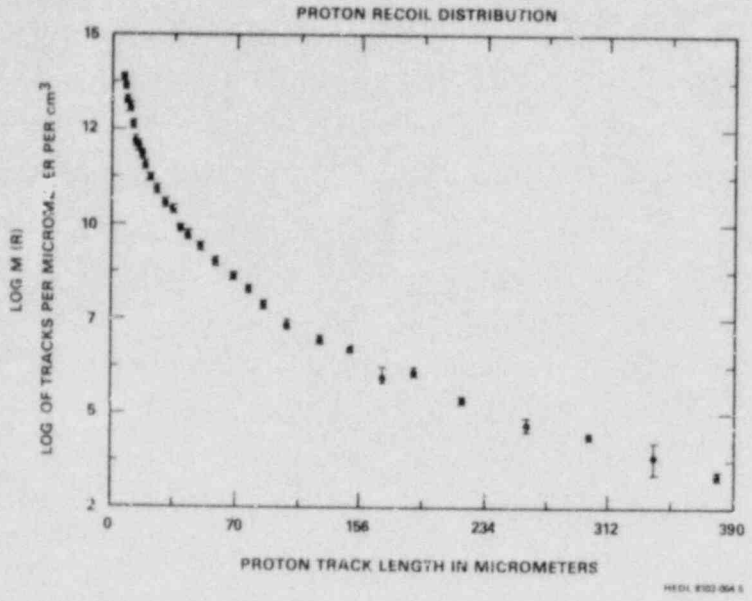

FIGURE HEDL-2. Proton Range Distribution Observed with the ESP System in a Reactor Core Irradiated Emulsion. Neg P10274-4

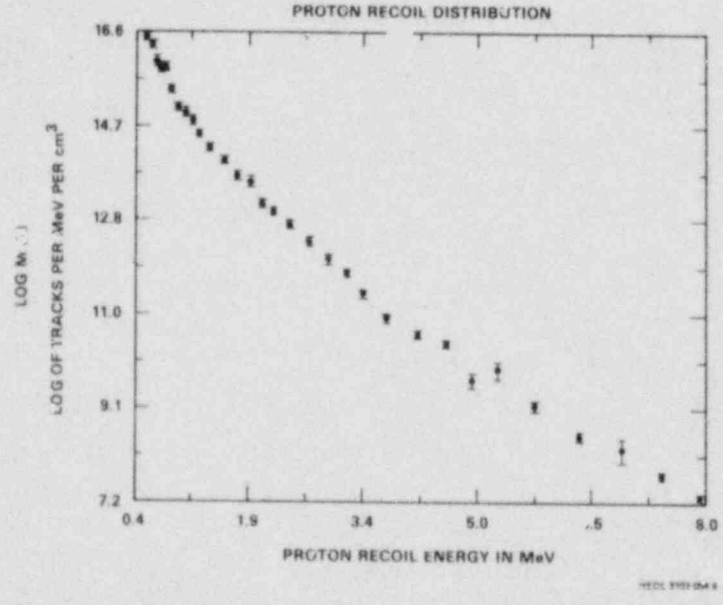

FIGURE HEDL-3. Proton-Recoil Energy Distribution Obtained from Transformation of the Proton Range Distribution (displayed in Figure HEDL-2). Neg P10274-5

$$M(E_T) = n_p \cdot t \int_{E_T}^{\infty} \frac{\sigma_{np}(E)\overline{\phi(E)}}{E} dE \qquad (2)$$

In fact, Eq. (1) is based upon this integral relation between the neutron spectrum and the observed proton spectrum. Using Eq. (2) together with the "first-guess" neutron spectrum, the best available fit to the observed M(E) data can be obtained.

For example, the "first-guess" neutron spectrum obtained in earlier emulsion work is fit reasonably well by the parametric form:

$$[\overline{\phi_1(E)}t] = Ae^{-aE} + BE^{1/2}e^{-bE} \quad , \qquad (3)$$

where the zero order values of the parameters A, a, B, and b are obtained from the "first-guess" neutron spectrum. Using this form of the "first-guess" neutron spectrum in Eq. (2), computer codes for parametric fitting, such as FATAL or VMM,[15] are then used to obtain parameter values that give the best overall fit of the observed proton spectrum, M(E).

Some general remarks about this procedure are warranted. For in-situ reactor environments, structure can exist in the neutron spectrum. In fact, data obtained with proton-recoil proportional counters reveal that such spectral structure can extend up to approximately 1 MeV in LWR-PV environments.[7] However, the spectral detail available with emulsions is considerably less, since it is impractical to amass counting statistics anywhere near that attained with proportional counters. Moreover, adequate energy resolution must exist in order to observe such spectral structure. The energy resolution available with emulsions is chiefly limited by uncertainty in track length measurements. This energy resolution decreases from approximately 80 keV to 50 keV (FWHM) with increasing neutron energy in the region from 0.4 MeV to 1.0 MeV.

Owing to the limited counting statistics and broader energy resolution normally attained with NRE, unfolded neutron spectra can not possess such detailed structure. Indeed, one can only expect to produce spectra that are suitably averaged over such structural detail. To this end, in working with emulsion data, one assumes the neutron spectrum is a structureless (monotone) continuum. Since we have denoted this monotone continuum by $\overline{\phi(E)}$, it follows from this assumption together with Eq. (2) that the proton spectrum is also a monotone continuum. It is within this context that the method described above can produce refined neutron spectra.

Hence, introduction of a parametric form to describe the neutron spectrum provides a monotone neutron continuum, wherein structure has been averaged out. However, the particular parametric form chosen need not have physical meaning. Therefore, physical meaning need not be ascribed to the parameters so introduced. In this context, the parametric form is merely a convenient means that can be justifiably introduced to obtain refined spectral results.

Operation in the Integral Mode -- Two different integral relationships can be established using proton-recoil emulsion data. These integral reaction rates can be obtained with roughly an order of magnitude reduction in scanning effort. Consequently, this integral mode is an important complementary alternative to the customary differential mode of emulsion spectrometry. The integral mode can be applied over extended regions, e.g., perhaps up to as many as 10 in-situ locations can be covered for the same scanning effort expended for a single differential measurement. Hence, the integral mode is especially advantageous for dosimetry applications that require extensive spatial mapping, such as exist in LWR-PV environments.

The first integral relationship follows directly from Eq. (2). The integral in Eq. (2) can be defined as:

$$I(E_T) = \int_{E_T}^{\infty} \frac{\sigma(E)}{E} \phi(E)\, dE \quad . \tag{4}$$

Here $I(E_T)$ possesses units of protons/(MeV-s) per hydrogen atom. Clearly $I(E_T)$ is a function of the lower proton energy cut-off used for analyzing the emulsion data. Using Eq. (4) in Eq. (2), one finds the integral relation:

$$I(E_T) = \frac{M(E_T)}{n_p \cdot t} \quad . \tag{5}$$

$I(E_T)$ is evaluated by using a least squares fit of the scanning data in the neighborhood of $E = E_T$. Alternatively, since:

$$M(E_T) = M(R_T) \frac{dR}{dE} \quad , \tag{6}$$

where $\frac{dR}{dE}$ is known from the proton range-energy relation in emulsions, one need only determine $M(R_T)$ in the neighborhood of $R = R_T$. Here $M(R)$ is the number of proton-recoil tracks per μ per cm^3 observed in the emulsion. Consequently, scanning efforts can be concentrated in the neighborhood of $R = R_T$ in order to determine $I(E_T)$. In this manner, the accuracy attained in $I(E_T)$ is comparable to the accuracy of the differential determination of $\phi(E)$, as based on Eq. (1), but with a significantly reduced scanning effort.

The second integral relation can be obtained by integration of the observed proton spectrum $M(E_T)$. From Eq. (2), one has:

$$\int_{E_{min}}^{\infty} M(E_T) dE_T = (n_p \cdot t) \cdot \int_{E_{min}}^{\infty} dE_T \cdot \int_{E_T}^{\infty} \frac{\sigma(E)}{E} \phi(E)\, dE \quad , \tag{7}$$

where E_{min} is the lower proton energy cutoff used in analyzing the emulsion data. Introducing into Eq. (7) the definitions:

$$\mu(E_{min}) = \int_{E_{min}}^{\infty} M(E_T)\, dE_T \quad , \tag{3}$$

and

$$J(E_{min}) = \int_{E_{min}}^{\infty} dE_T \int_{E_T}^{\infty} \frac{\sigma(E)}{E} \phi(E)\, dE \quad , \tag{9}$$

one has

$$J(E_{min}) = \frac{\mu(E_{min})}{n_p\, t} \tag{10}$$

Hence, the second integral relation, namely Eq. (10), can be expressed in a form analogous to the first integral relation, namely Eq. (5). Here $\mu(E_{min})$ is the integral number of proton-recoil tracks/cm^3 observed above an energy E_{min} in the emulsion. Consequently, the integral $J(E_{min})$ possesses units of tracks per second per hydrogen atom.

The integral $J(E_{min})$ can be reduced to the form

$$J(E_{min}) = \int_{E_{min}}^{\infty} \left(1 - \frac{E_{min}}{E}\right) \sigma(E)\phi(E)\, dE \tag{11}$$

In addition, by using Eq. (6), the observable $\mu(E_{min})$ can be expressed in the form

$$\mu(E_{min}) = \int_{R_{min}}^{\infty} M(R)dR \tag{12}$$

Hence, to determine the second integral relationship, one need only count proton-recoil tracks above $R = R_{min}$. Tracks considerably longer than R_{min} need not be measured, but simply counted. However, for tracks in the neighborhood of $R = R_{min}$, track length must be measured so that an accurate lower bound R_{min} can be effectively determined.

It is of interest to compare the differential energy responses available from these two integral relations. From Eqs. (4) and (11), one finds responses of the form $\sigma(E)/E$ and $(1 - E_{min}/E)\sigma(E)$ for the first and second integral relations, respectively. Note that the integrals $I(E_T)$ and $J(E_{min})$ possess different units, namely protons/MeV-s and protons/s, respectively. These two responses are compared in Figure HEDL-4, using a common cutoff of 0.5 MeV for both E_T and E_{min}.

Since these two responses are substantially different, simultaneous application of these two integral relations would be highly advantageous. This can be accomplished using the $J(E_{min})$ scanning mode with some additional emphasis placed on tracks in the neighborhood of $R = R_{min}$. As already noted, one must determine R_{min} for the $J(E_{min})$ integral. Hence, to determine $J(E_{min})$ one must simultaneously determine the behavior of $M(R)$ in the neighborhood of $R = R_{min}$. Therefore, one finds that the integral $I(E_T)$ can be determined for $E_T = E_{min}$ from the $J(E_{min})$ scanning mode, with only a slight increase in overall scanning effort.

Estimation of Uncertainties -- Sources of systematic uncertainty for absolute neutron spectrometry with NRE are summarized in Table HEDL-7. Since these sources of uncertainty are essentially independent, the quadrature uncertainty for all systematic effects comes to roughly 5%. It should be stressed that Table HEDL-7 does not include uncertainties arising from the in-situ irradiation such as exist in exposure time t as well as in absolute power level. For emulsion irradiations in the LWR-PVS, the uncertainty estimate for the exposure time is approximately 1% (1σ) whereas the uncertainty estimate for the absolute power level is approximately 6% (1σ) for emulsion irradiations carried out to date (see Tables HEDL-5 and HEDL-6).

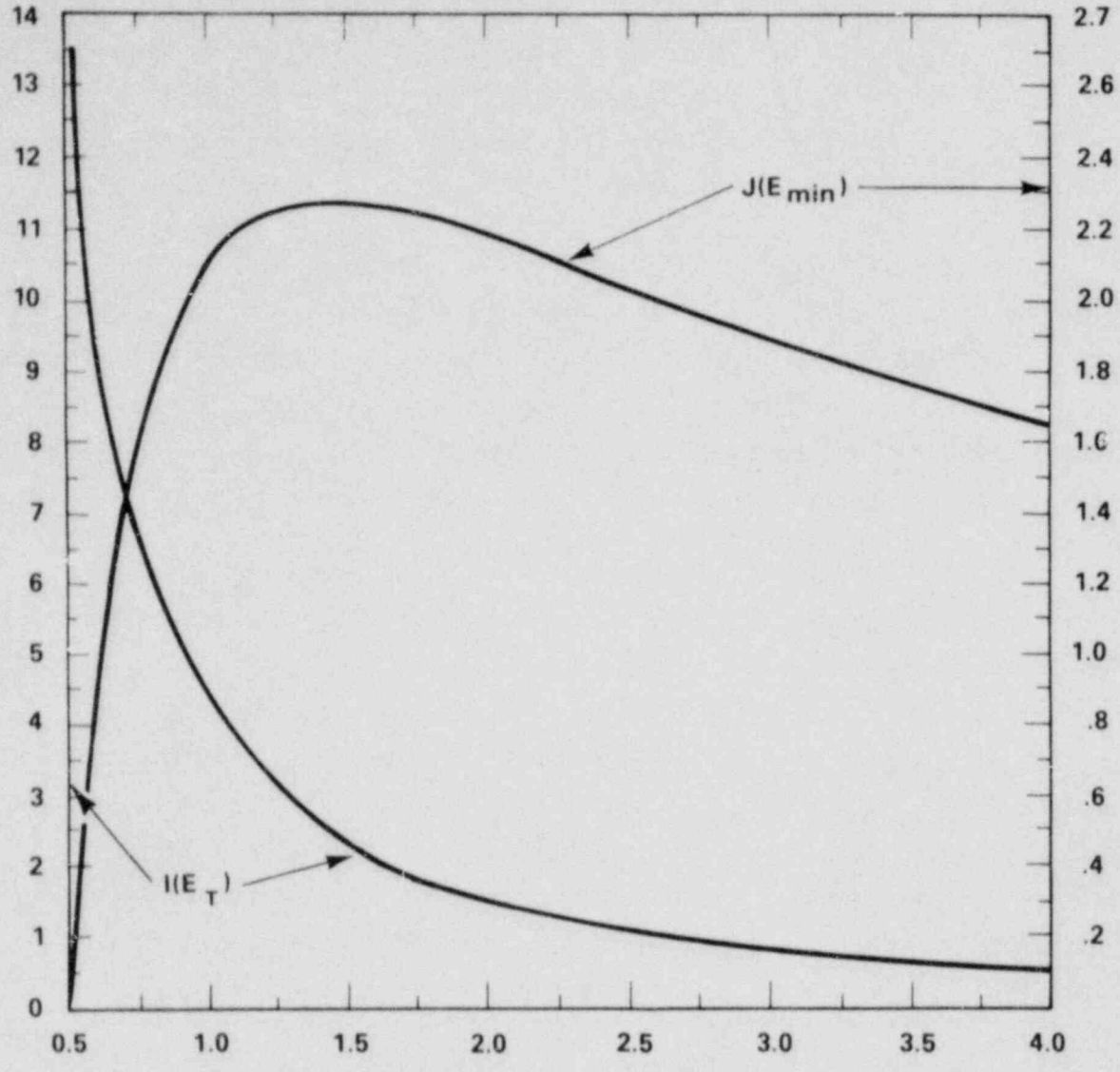

FIGURE HEDL-4. Response Factors for the Integral Reaction Rate $I(E_T)$ and $J(E_{min})$. Neg P11401-3

TABLE HEDL-7

UNCERTAINTY ESTIMATES FOR ABSOLUTE NEUTRON SPECTROMETRY WITH NUCLEAR RESEARCH EMULSIONS

Source of Uncertainty	Approximate Relative Uncertainty (1σ)
1. Proton range straggling	2%
2. Proton energy based on range-energy relation	2%
3. Range measurements	2%
4. Volume of emulsion scanned	2%
5. Hydrogen density in the emulsion	3%
6. Hydrogen $\sigma_{np}(E)$ cross section	1%

As already noted, owing to the dependence of the neutron spectrum upon the derivative of the observed proton spectrum [see Eq. (1)], observation of about 10^4 tracks will yield a statistical uncertainty ranging from approximately 2 to 10% (1σ) with an associated energy resolution of approximately 10%. Consequently the overall uncertainty for absolute neutron spectrometry with NRE in the LWR-PVS varies from approximately 6 to 13% (1σ).

For the integral mode, the systematic uncertainties given in Table HEDL-7 apply as do the irradiation uncertainties in exposure time and absolute power level. The uncertainties in the absolute reaction rates $I(E_T)$ and $J(E_{min})$ follow from Eqs. (5) and (10), respectively. The relative variances for these absolute reaction rates are given by:

$$\left(\frac{\delta I}{I}\right)^2 = \left(\frac{\delta M}{M}\right)^2 + \left(\frac{\delta n_p}{n_p}\right)^2 + \left(\frac{\delta t}{t}\right)^2 \quad , \tag{13}$$

and

$$\left(\frac{\delta J}{J}\right)^2 = \left(\frac{\delta \mu^2}{\mu}\right) + \left(\frac{\delta n_p}{n_p}\right)^2 + \left(\frac{\delta t}{t}\right)^2 \qquad (14)$$

The relative variance $(\delta M/M)^2$ in Eq. (13) stems from a least squares fit of the proton spectrum in the neighborhood of $E = E_T$. Hence this relative variance depends upon counting statistics as well as the systematic uncertainties (1-4) of Table HEDL-7. For the LWR-PVS irradiations, the uncertainty $(\delta M/M)$ is estimated to be approximately 5% (1σ), assuming only 10^3 tracks have been observed in the integral mode. Using Eq. (13) together with the aforementioned absolute power uncertainty of 6% (1σ), one finds an overall uncertainty in $I(E_T)$ of approximately 8.5% (1σ).

Slightly larger uncertainties arise for the absolute reaction rate $J(E_{min})$. The relative variance $(\delta\mu/\mu)^2$ in Eq. (14) also depends upon counting statistics and the systematic uncertainties (1-4) of Table HEDL-8. Again assuming that about 10^3 tracks have been observed in the integral scanning mode, the estimated value of $(\delta\mu/\mu)$ is approximately 9% (1σ). Using this estimate in Eq. (14) together with the aforementioned 6% (1σ) uncertainty in absolute power level, one finds an overall uncertainty in $J(E_{min})$ of approximately 12% (1σ).

Expected Accomplishments

Emulsions irradiated in the 8/7, 12/13, and 4/12 configurations of the LWR-PVS will be scanned in the integral mode. These emulsions based absolute reaction rates will be used as input for neutron spectral adjustment codes.

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OAK RIDGE NATIONAL LABORATORY
(ORNL)

A. LIGHT WATER REACTOR PRESSURE VESSEL (LWR-PV) BENCHMARK FACILITIES (PCA, ORR-PSF, ORR-SDMF) at ORNL

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Objectives

In order to serve as benchmarks, the neutron fields at PCA, ORR-PSF, and BSR-SDMF need to be known and controlled within sufficiently narrow uncertainty bounds. To achieve this objective, extensive measurements are combined with neutron physics calculations. Statistical uncertainty analysis and unfolding techniques are used to determine uncertainty bounds. The results of this task will have a direct impact in the preparation of ASTM Standards for Surveillance of Nuclear Reactor Pressure Vessels. The objectives of these benchmark fields are:

1) PCA (in operation)--to validate and improve neutron transport calculations and dosimetry techniques in LWR-PV environments;
2) ORR-PSF (in operation)--to obtain reliable information from dosimetry measurements and neutron transport calculations and to correlate the spectral parameters with structural changes in the pressure vessel;
3) ORR-SDMF--to investigate results of current surveillance capsules so that dosimetry methods applied by vendors and service laboratories can be:
   a) validated and certified;
   b) improved by development of supplementary experimental data; and
   c) evaluated in terms of actual uncertainties.

The three facilities provide direct support for the improvement and validation of the following ASTM Standards:

1) Methods of Surveillance and Correlation Practices:
   a) Analysis and Interpretation of Nuclear Reactor Surveillance Results,
   b) Analysis and Interpretation of Nuclear Reactor Surveillance Results,
   c) Surveillance Tests for Nuclear Reactor Vessels,
   d) Extrapolating Reactor Vessel Surveillance Dosimetry Results,
   e) Characterizing Neutron Exposures in Ferretic Steels in Terms of Displacements Per Atom, Including ASTM ENDF/A DPA File, and
   f) Damage Correlation for Reactor Vessel Surveillance.

2) Supporting Methodology Guides:
   a) Application of Multiple Sensor Flux Fluence Spectral Determination Codes,
   b) Application of ASTM/ENDF/A Cross Section and Error File,
   c) Sensor Set Design and Irradiation for Reactor Vessel Surveillance,
   d) Application of Neutron Transport Methods for Reactor Vessel Surveillance, and
   e) Benchmark Testing of Reactor Neutron Dosimetry.

3) Sensor Measurements Methods:
   a) Analysis of Radiometric Monitors for Reactor Vessel Surveillance,
   b) Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Vessel Surveillance,
   c) Analysis of Helium Accumulation Flux/Fluence (HAFM) Monitors for Reactor Vessel Surveillance,
   d) Analysis of Damage Monitors for Reactor Vessel Surveillance, and
   e) Analysis of Temperature Monitors for Reactor Vessel Surveillance.

A.1. Pressure Vessel Benchmark Facility for LWR Reactor Physics and Neutron and Gamma Dosimetry Improvement and Validation (PCA)

Summary

Documentation of the NRC Computational "Blind Test" results and related data was completed,[3] and a question regarding transverse leakage treatments by "Blind Test" participants was resolved based on Monte Carlo calculations.[4]

Accomplishments and Status

Compilation and evaluation of the PCA calculational and experimental results is under way for the preparation of a NUREG report in early 1981. Monte Carlo results performed at ORNL[4] substantiated the fact that the under-estimation of fluxes in the transport calculations versus measurements in the pressure vessel simulator is not the result of leakage correction factors.

Expected Accomplishments in the Next Reporting Period

Documentation for the NUREG report on calculations and dosimetry for the LWR-PV Surveillance Dosimetry Improvement Program should be completed.

A.2. Pressure Vessel Benchmark Facility for LWR Metallurgical Testing of Reactor Pressure Vessel Steels (ORR-PSF)

Summary

Irradiation history and thermal data for the simulated PVC were processed for data through Dec. 1980. Analysis of the metallurgical specimens and dosimetry for SSC-1 is continuing.

Accomplishments and Status

Characterization data for the surveillance and pressure vessel capsules are presented in Tables 1 and 2, respectively. Note that the surveillance capsule results represent a final summary and that results for the pressure vessel capsule are intermediate but are cumulative through Dec. of 1980. It is apparent from the histograms and the standard deviations associated with the average temperature of each thermocouple that specimen temperatures are maintained close to the desired 288°C.

Analysis of the neutron characterization and testing of metallurgical specimens in the SSC-1 capsule is continuing. Counting data from gradient wires in the SSC-1 were analyzed to obtain preliminary estimates of fluxes greater than 1.0 MeV at the locations of the Charpy specimens. Results are given in Tables 3 and 4.

Expected Accomplishments in the Next Reporting Period

Additional PVS capsule irradiation data will be processed and the analysis of SSC-1 metallurgical and dosimetry related data will continue.

A.3. Surveillance Dosimetry Measurement Benchmark Facility (SDMF) for Validation and Certification of Neutron Exposures from Power Reactor Surveillance

Summary

A committee to review the SDMF has recommended that the facility be located at the ORR instead of the BSR. Current plans call for adapting the ORR-PSF to accomplish the SDMF tasks after completion of the PSF irradiations in FY-82.

Table 1. Irradiation and Temperature Distribution Data from April 30-June 23, 1980.

Data for PSF Specimen Set SSC-1
Hours of Irradiation Time = 1075.29
Magawatt Hours of Irradiation = 32017.57

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE 1	20.84	283.91	770.55	0.00	0.00	281.25	2.43
TE 2	15.77	4.70	1045.66	9.17	0.00	291.15	1.64
TE 3	17.82	3.12	1054.34	0.00	0.00	288.49	1.74
TE 4	7.11	9.33	364.74	694.12	0.00	295.39	3.03
TE 5	16.15	3.29	1049.00	6.83	0.00	289.70	1.87
TE 6	8.25	10.78	977.42	78.84	0.00	292.32	1.99
TE 7	340.97	573.48	160.86	0.00	0.00	275.69	6.29
TE 8	19.50	7.78	1047.84	0.17	0.00	286.18	1.82
TE 9	10.81	8.75	702.94	352.80	0.00	295.18	1.87
TE 10	42.13	875.54	157.63	0.00	0.00	275.30	5.17
TE 11	20.05	131.04	924.21	0.00	0.00	281.90	1.40
TE 12	19.23	106.34	949.71	0.00	0.00	283.51	2.84
TE 13	18.63	5.61	1010.84	40.21	0.00	289.42	2.70
TE 14	19.20	2.80	698.51	354.77	0.00	294.82	2.58
TE 15	19.21	5.31	1050.77	0.00	0.00	287.64	1.62
TE 16	23.64	11.49	1040.16	0.00	0.00	285.61	1.69
TE 17	19.20	9.38	1046.09	0.00	0.00	287.05	1.43
TE 18	20.65	11.53	1043.11	0.00	0.00	288.24	2.41
TE 19	19.82	15.85	1039.61	0.00	0.00	284.07	1.73
TE 20	27.85	46.31	1001.14	0.00	0.00	283.61	2.37

Table 2. Cumulative Characterization Data for the Pressure Vessel Capsule Through December 31, 1980.

Data for PSF Specimen Set OT
Hours of Irradiation = 4174.92
Megawatt Hours of Irradiation = 118294.17

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE 101	53.13	16.19	4084.22	21.38	0.00	289.64	1.84
TE 102	52.20	11.91	4068.14	42.66	0.00	291.31	1.38
TE 103	51.70	10.68	4112.54	0.00	0.00	289.13	1.03
TE 104	47.63	9.38	4109.48	8.44	0.00	291.88	0.95
TE 105	51.29	13.36	4110.30	0.00	0.00	285.58	1.15
TE 106	47.41	10.05	4117.45	0.00	0.00	288.72	1.03
TE 107	52.64	338.75	3783.55	0.00	0.00	282.06	1.44
TE 108	55.91	13.25	4096.66	9.06	0.00	289.42	1.64
TE 109	55.93	14.24	4096.95	7.78	0.00	288.96	1.66
TE 110	51.62	12.29	4098.08	12.95	0.00	290.23	1.11
TE 111							
TE 112							
TE 113	46.48	9.66	4116.72	0.04	2.00	290.42	1.64
TE 114	58.07	14.31	4102.51	0.00	0.00	288.74	[illegible]
TE 115							
TE 116	55.68	9.82	4109.42	0.00	0.00	289.97	[illegible]
TE 117	52.08	10.65	4106.39	5.29	0.50	291.35	0.95
TE 118	54.70	15.82	4104.46	0.00	0.00	286.33	1.08
TE 119	51.77	13.70	4109.45	0.00	0.00	286.64	1.05
TE 120	54.78	189.27	3930.92	0.00	0.00	282.93	1.54

Data for PSF Specimen Set 1/4 T
Hours of Irradiation Time = 4174.92
Megawatt Hours of Irradiation = 118294.17

TE 201	53.00	14.22	4105.05	2.66	0.00	290.28	1.50
TE 202	53.27	13.73	4107.74	0.17	0.00	288.89	1.05
TE 203	51.90	11.21	4111.84	0.00	0.00	288.60	1.08
TE 204	49.99	11.54	4113.08	0.33	0.00	289.40	0.87
TE 205	49.82	18.52	4106.61	0.00	0.00	286.32	1.04
TE 206	47.93	16.76	4110.24	0.00	0.00	286.84	0.94
TE 207	51.61	44.59	4078.73	0.00	0.00	282.94	1.07
TE 208	54.07	11.04	4108.97	0.83	0.00	288.12	1.44
TE 209	54.45	14.83	4105.64	0.00	0.00	288.67	1.34
TE 210	53.15	21.15	4100.68	0.00	0.00	286.73	0.83
TE 211	55.31	25.02	4094.61	0.00	0.00	284.83	0.82
TE 212	47.40	7.48	4120.04	0.00	0.00	290.84	1.05
TE 213	47.76	8.01	4119.13	0.00	0.00	289.60	1.16
TE 214	54.62	10.43	4109.85	0.00	0.00	290.47	0.99
TE 215	54.85	14.73	4105.31	0.00	0.00	287.49	0.74
TE 216	53.97	11.09	4109.88	0.00	0.00	287.85	0.79
TE 217	50.49	8.85	4115.59	0.00	0.00	289.73	0.89
TE 218	50.41	14.48	4108.07	2.00	0.00	286.76	1.07
TE 219	48.95	11.64	4114.36	0.00	0.00	286.99	0.94
TE 220	50.03	104.86	4020.05	0.00	0.00	285.00	1.36

Table 2. (Cont'd)

Data for PSF Specimen Set 1/2 T
Hours of Irradiation Time = 4174.92
Megawatt Hours of Irradiation = 118294.17

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE 301	51.66	7.32	4076.80	39.18	0.00	290.09	1.10
TE 302	53.58	11.88	4109.48	0.00	0.00	286.38	0.87
TE 303	50.65	9.51	4114.77	0.00	0.00	287.29	0.96
TE 304	47.90	9.29	4117.15	0.58	0.00	291.08	0.76
TE 305	47.59	10.03	4117.30	0.00	0.00	287.65	0.95
TE 306	49.06	14.41	4111.44	0.00	0.00	286.61	0.87
TE 307							
TE 308	53.91	7.92	4113.09	0.00	0.00	289.18	1.15
TE 309	55.29	10.46	4109.19	0.00	0.00	287.58	0.90
TE 310	55.78	24.03	4095.14	0.00	0.00	285.55	1.03
TE 311	55.95	22.99	4096.03	0.00	0.00	285.90	1.16
TE 312	48.89	7.07	4118.79	0.17	0.00	288.64	0.91
TE 313	48.84	7.08	4117.35	1.67	0.00	289.96	1.05
TE 314	56.02	10.28	4108.62	0.00	0.00	289.11	0.95
TE 315	56.74	11.20	4107.00	0.00	0.00	285.13	0.90
TE 316	55.02	4.38	4115.54	0.00	0.00	287.57	0.77
TE 317	50.94	7.68	4116.31	0.00	0.00	290.87	0.89
TE 318	50.54	7.70	4116.68	0.00	0.00	289.30	0.99
TE 319	52.62	16.36	4105.94	0.00	0.00	285.26	0.83
TE 320	50.99	12.05	4111.88	0.00	0.00	287.53	1.34

Table 3. PSF-SSC-1 Fe Gradient [$^{54}Fe(n,p)^{54}Mn$] Analytical Results from Cut Segments.

Wire No. 1		Wire No. 2		Wire No. 3		Wire No. 4	
Z(mm)	CPS/mg @ EOI	Z(mm)	CPS/mg @ EOI	Z(mm)	CPS/mg @ EOI	Z(mm)	CPS/mg @ EOI
+123	14.27	-	-	+117.7	9.292	+123	9.064
+105.7	15.22	+105.7	15.53	+114.1	9.375	+114.0	9.217
+ 80.3	15.65	+ 80.3	16.04	+ 92.3	9.545	+105.7	9.597
+ 54.9	16.15	+ 54.9	16.48	+ 66.9	9.903	+ 80.3	9.968
+ 29.5	16.55	+ 29.5	16.76	+ 41.5	10.17	+ 54.9	10.16
+ 4.1	16.55	+ 4.1	16.76	+ 16.1	10.17	+ 29.5	10.26
- 21.3	16.37	- 21.3	16.62	- 9.3	10.08	+ 4.1	10.45
- 46.7	15.98	- 46.7	16.38	- 34.7	10.09	- 21.3	10.26
- 72.1	15.59	- 72.1	15.81	- 60.1	9.714	- 46.7	10.02
- 97.5	15.01	- 97.5	15.03	- 85.5	9.380	- 72.1	9.590
-119.8	14.27	-119.8	14.14	-110.9	8.899	- 97.5	9.259
				-118.9	8.542	-119.8	8.614

x S 103.7 ± 0.4 mm N 103.7 ± 0.4 mm x S 103.7 ± 0.4 mm N 103.7 ± 0.4 mm

y 221.3 $^{+0.5}_{-1.1}$ Y 221.3 $^{+0.5}_{-1.1}$ y 244.5 $^{+1.1}_{-0.5}$ y 244.5 $^{+1.1}_{-0.5}$

Distance between wires x Wires 1 and 2, 3 and 4 207.4 mm ± 0.8 mm

y Wires 1 and 3, 2 and 4 23.2 mm $^{+2}_{-0.8}$ mm

Z dimensions are referenced to the SSC-1 assembly centerline (R C_L) rather than core centerline (C C_L). The R C_L was given as 50.8 mm below C C_L.

Table 4. PSF-SSC-1 [$^{54}Fe(n,p)^{54}Mn$].

Radial (X) Gradient from Plot of Analytical Data			Radial (y) Gradient from Plot of Analytical Data		
Z	Ratio 2/1 (NF/SF)	Ratio 4/3 (NR/SR)	Z	Ratio 1/3 (SF/SR)	Ratio 2/4 (NF/NR)
+120	1.0380*	0.9967*	+120	1.5978*	1.6634*
+100	1.0371	1.0148	+100	1.5973	1.6319
+ 80	1.0255	1.0211	+ 80	1.6069	1.6160
+ 60	1.0210	1.0220	+ 60	1.6092	1.6078
+ 40	1.0180	1.0210	+ 40	1.6098	1.6065
+ 20	1.0121	1.0176	+ 20	1.6168	1.6028
+ 0	1.0132	1.0143	0	1.6142	1.6059
- 20	1.0153	1.0104	- 20	1.6134	1.6168
- 40	1.0177	1.0120	- 40	1.6167	1.6265
- 60	1.0207	1.0113	- 60	1.6209	1.6371
- 80	1.0146	1.0108	- 80	1.6283	1.6376
-100	1.0137	1.0022	-100	1.6349	1.6604
-120	0.9944	1.0100	-120	1.6674	1.6404
	1.0169 ± 0.0099 1σ (0.97%)	1.0134 ± 0.0058 1σ (0.52%)		1.6197 ± 0.0179 1σ (1.11%)	1.6233 ± 0.0184 1σ (1.13%)

*Calculated from extrapolated axial plots.

B. ASTM STANDARDS FOR SURVEILLANCE OF NUCLEAR REACTOR PRESSURE VESSELS

Objectives and Summary

Efforts relative to ASTM Standards have involved a critical review of a Standard Practice for the Analysis of Charpy Impact Data and the update of the ASTM Guide on the Application of Neutron Transport Methods.

Accomplishments and Status

A critical review of a proposed new Standard Practice for the Analysis of Charpy Impact Data has been performed by F. W. Stallmann. A new procedure for the analysis of Charpy data has been developed and will be readied for publication during the next reporting period.

The ASTM Standard Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance was updated and thoroughly reviewed by several ORNL personnel. It was also forwarded to appropriate ASTM E10.05 committee members for review prior to an E10.05 meeting scheduled for Feb. 1981 in Reno, Nevada. Also, a working paper on the assessment of propagation of uncertainties for the LWR-PV surveillance program was prepared for the ASTM E10.05.01 task group at the Reno meeting.

Expected Accomplishments in the Next Reporting Period

The ASTM Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance will be updated based on comments from the ASTM E10.05 meeting in Reno, Nevada in Feb. 1981.

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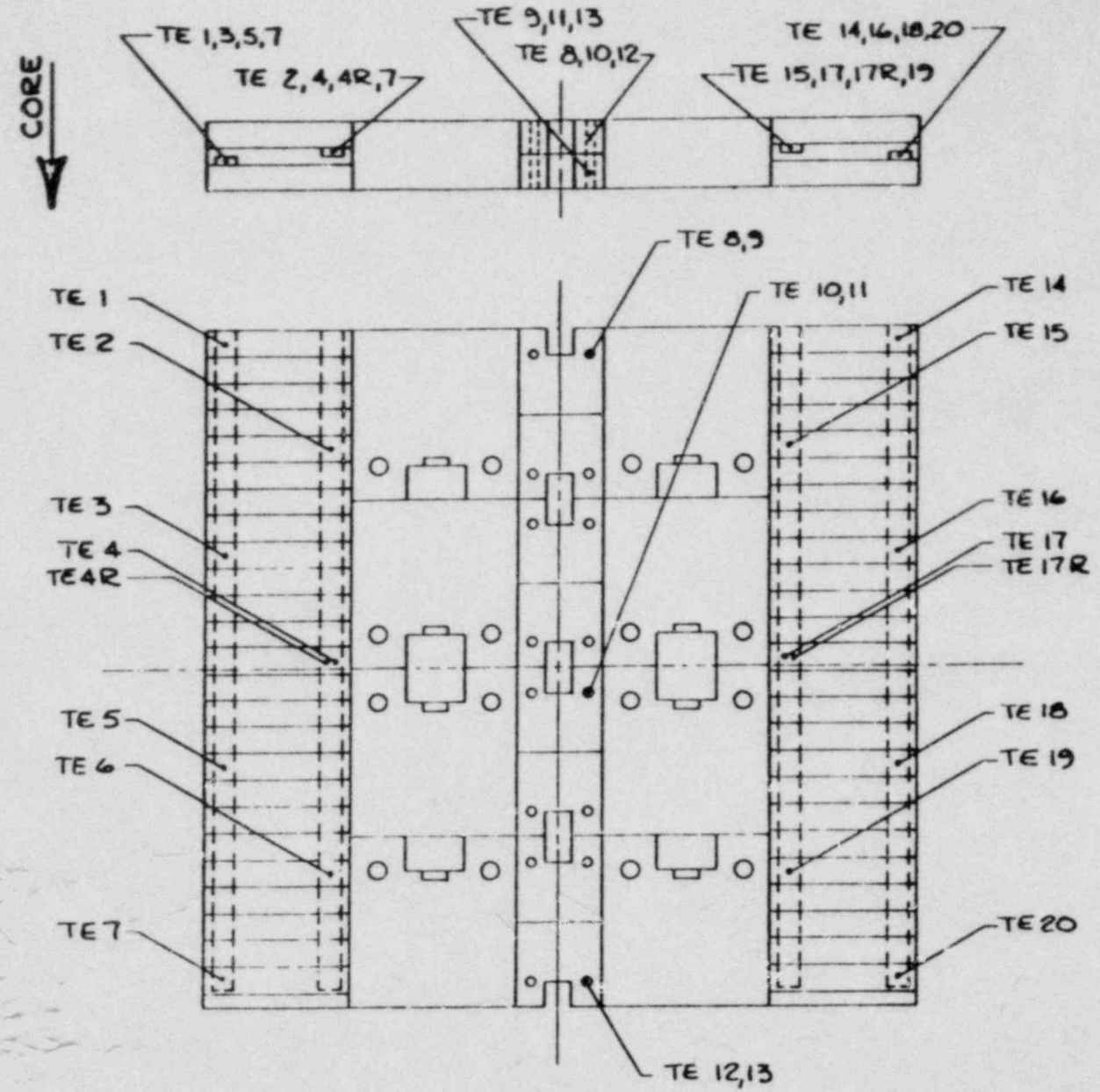


Figure 1. Thermocouple Locations in SSC-1.

APPENDIX
CEN/SCK REPORTS

A number of CEN/SCK reports have been omitted from past volumes of this series due to problems with publishing schedules. The reports (photocopies of originals) are included here as an appendix so that they may be available for reference purposes.

A. OVERVIEW OF THE CEN-SCK LIGHT WATER REACTOR DOSIMETRY - METALLURGY PRESSURE VESSEL SURVEILLANCE IMPROVEMENT PROGRAM

A. Fabry, J. Debrue and P. VanAsbroeck

Objective

Summarize the CEN-SCK program status and the directions of efforts in the frame of the current cooperation and coordination with the NRC-sponsored program and with the KFA (Jülich) program.

Summary

The status and progress of the CEN-SCK program are reviewed along its three Tasks: Metallurgy Data Base Standardization; In-vessel Extrapolation of Surveillance Dosimetry Data; Dosimetry Assessment and Standardization (LWR Pressure Vessel Irradiation Surveillance Dosimetry Quarterly Progress Report, July - September 1977, NUREG/CR-0038 CEN-SCK-1 to 11)

Highlighted are:

- the final selection of a national ferritic steel forging procured and documented by the COCKERILL Nuclear Division and considered valuable as a reference alloy for the program
- significant contribution to the coordination and preparation, completed, of metallurgy - dosimetry irradiations in FRJ-2, FRJ-1 (Jülich) and BR3 (Mol)
- the efforts in participating to the validation of the ASTM Transport Theory Guide by experimental and analytical work under course at the MOL Iron Shells and the PCA Physics Benchmark Neutron Fields

Included are introductory comments for a section which provides information related to power plant surveillance dosimetry in Belgium and is aimed as an interface with the present research and development program.

Accomplishments and Status

I. Administration

Significant and sustained Belgium contribution to the definition, coordination and implementation of the Interlaboratory LWR Pressure Vessel Surveillance Improvement Program currently encompasses the efforts of many CEN-SCK staff members at MOL, the temporary assignment of some CEN-SCK experts at ORNL, the

participation to frequent meetings at ORNL, HEDL, NBS, KFA (Jülich), CEN-SCK (MOL) and COCKERILL (Liege), as well as membership to ASTM Committee E-10 for related work on ASTM standards, guides and practices. Reporting in this series is planned on a semi-annual basis. Covered here is the period January 1 to June 30, 1978.

II. Metallurgy Data Base Standardization

(Emphasis of this Task is on damage correlation between different environments for selected pressure vessel ferritic steels).

In cooperation with experts of the COCKERILL Nuclear Division, final identification and procurement has been made of a SA508 cl. 3 steel forging which, while primarily of national interest (COCKERILL Integral Vessels, new Belgium power plants under implementation or design for the DOEL and TIHANGE sites), does feature such chemical, microstructural and mechanical properties and homogeneity as to render it a useful reference for comparison with the steels investigated by other program participants. The emphasis is to assess the neutron fluence spectrum and flux level sensitivity of radiation-induced embrittlement, relatively to HSST plate 02 or plate 03 material (SA533B cl. 1 plate) which has been provided by NRC/ORNL for the needs of the cooperation program. Available overall documentation for the unirradiated SA508 reference forging has been disseminated to the participants (Minutes of the June 20, 1978 CEN-SCK/COCKERILL Meeting on Fracture Toughness Characterization of a Reference Belgium LWR Pressure Vessel Steel SA508 cl. 3 Forging, prepared by A. Fabry, reference AF/eu 380/78-21, July 5, 1978).

Suitable space has been allocated by NRC to CEN-SCK for the irradiation of Charpy-V, tensile and small metallurgical specimens (TEM, compression discs) in the ORR-PSF high flux pressure vessel benchmark. The counterpart CEN-SCK commitment encompasses active contribution to the preparation and coordination of interlaboratory PSF dosimetry, to the definition of the irradiation configuration and strategy, as well as an extensive neutronic characterization effort at the PCA low flux pressure vessel benchmark (see below and sections B1 to B4).

In cooperation with KFA (Jülich) and HEDL, joint German-U.S.-Belgium metallurgy-dosimetry irradiations in the FRJ-2 (D_2O moderated) and FRJ-1 (H_2O moderated) test reactors have been thoroughly prepared (section D). This is aimed at providing: 1) initial standardization between the three programs 2) a tie with the IAEA coordinated program 3) an independent, early look into the neutron spectrum sensitivity of some of the steels selected for forthcoming PSF irradiations.

The improved metallurgy-dosimetry surveillance practices under development, part'y by the present interlaboratory program, must be tested for applicability and optimization (cost versus benefit) in the more "real world" of power plants. The BR3 Test Regions program (section E), involving CEN-SCK, COCKERILL, NRC/HEDL/AI and EPRI, reflects such consideration to a large degree. Irradiations in the core 4B configuration in- and out-vessel locations have been planned and prepared very carefully, but are delayed by approximately one year (start during Spring 1979).

III. In-Vessel Extrapolation of Surveillance Dosimetry Data

(Emphasis of this Task is on improvement and validation of transport theory analysis as applied to LWR surveillance).

The Mol-II and Cavity Uranium-235 Fission Spectrum Standard Neutron Fields, well characterized and documented in the open literature, are used for sensor calibration and referencing (sections B1, B2).

The Mol Iron-Shell Reference Neutron Fields benchmark (section C) is an important complement to PCA for the validation of the "Transport Theory Guide": as calculations of this one-dimensional facility do not entail modeling uncertainties, ideal testing and adjustment of iron cross section data is possible in the energy range where spectral fine structure of neutron penetration through steel most significantly impacts on pressure vessel embrittlement projection uncertainties.

The CEN-SCK characterization efforts for PCA encompass passive radiometric, active fission chamber and ^{6}Li (n,α) neutron spectrometry measurements as well as one- and two-dimensional transport theory calculations (sections B1 to B4). Program management support is also provided and has been noticeably successful in helping at the definition of a PCA clean core configuration and in establishing the need as well as an adequate approach for absolute core power distribution measurements.

IV. Dosimetry Assessment and Standardization

The bulk of the efforts related to this Task currently focuses on interlaboratory comparisons of "established" and "advanced" techniques for surveillance fluence spectrum monitoring (in particular sections D and E).

The outline of a practice for "Damage Monitors" has been submitted to the ASTM E10-05 Steering Subcommittee. Validation experiments are in progress.

V. Interface with Power Plant Surveillance

The usefulness of the present research and development program is largely enhanced if it is given the opportunity to impact on dosimetry and metallurgy surveillance practices in power reactors. Responsibility for the mechanical testing and dosimetry measurements for surveillance capsules irradiated in Belgian power plants has been assigned to CEN-SCK. The laboratory is thus in good position to offer constructive proposals and dialogue for the improvement of surveillance capsule design, exposure strategies, and so on. Although this is not part of the Tasks committed as belonging to the interlaboratory program (LWR Pressure Vessel Irradiation Surveillance Dosimetry Quarterly Progress Report, July-September 1977, NUREG/CR-003R (CEN-SCK-1 to 11), documentation is (section F) and will be provided, as appropriate, of dosimetry and metallurgy information considered pertinent in encouraging "interface" activities and interaction.

B. PCA PRESSURE VESSEL BENCHMARK CHARACTERIZATION:

1. Radiometric Reaction Rate Measurements

A. Fabry, F. Cops

Objective

Measure reaction rates and spectral indices for a selected set of radiometric sensors exposed in the PCA access channels within the water, pressure vessel steel and void box environments.

Status and Summary of Accomplishments

The selection and preparation of sensors has been completed. Encompassed are the reactions $^{103}Rh(n,n')^{103m}Rh$, $^{115}In(n,n')^{115m}In$, $^{58}Ni(n,p)^{58}Co$, $^{32}S(n,p)^{32}P$(*), $^{56}Fe(n,p)^{56}Mn$, $^{27}Al(n,\alpha)^{24}Na$, $^{197}Au(n,\gamma)^{198}Au$, $^{55}Mn(n,\gamma)^{56}Mn$. Adequate PCA sensor boxes, trays and irradiation holders have been designed, fabricated and shipped to ORNL. Calibration of the gamma-ray, X-ray and β-ray spectrometers needed for these measurements has been initiated. A crucial component of these calibrations involves exposures in the CEN-SCK and NBS Cavity Uranium-235 Fission Spectrum Standard Neutron Fields.

(*) This sensor type has been prepared by L. S. Kellogg, HEDL.

B. PCA PRESSURE VESSEL BENCHMARK CHARACTERIZATION:

2. Miniature Fission Chamber Measurements

A. Fabry

Objectives

Measure reaction rates and spectral indices for the $^{237}Np(n,f)$, $^{238}U(n,f)$ and $^{235}U(n,f)$ reactions in the PCA access channels within the water, pressure vessel steel and void box environments, relatively to the MOL Standard Neutron Fields.

Assess cavity neutron field perturbations in the absolute measurements by means of the NBS double gas-flow fission chambers.

Determine PCA axial bucklings and investigate their space- and energy-dependence.

Status and Summary of Accomplishments

A set of five commercially-available, high-pressure gas, sealed miniature fission chambers (diameter 5mm, active length 10mm) has been procured; redundancy is provided for ^{237}Np and ^{238}U. The chambers have been tested and their calibration by exposure in the CEN-SCK Cavity Uranium-235 Fission Spectrum and Mol-IIC Standard Neutron Fields is in progress. Adequate irradiation plugs and manipulation tools for PCA measurements have been prepared. All equipment has been shipped to ORNL.

B. PCA Pressure Vessel Mock-up Characterization :

3. ^{6}Li-Spectrometry

G. DE LEEUW-GIERTS, S. DE LEEUW, F. COPS

Objective

Determine the neutron spectrum tentatively from about 50 keV up to several MeV by means of the Li-6 method, at least at four positions, on Face of the Pressure Vessel, 1/4 , 1/2 and 3/4 thickness into the Pressure Vessel.

Status

Scoping experiments will be performed in the PCA in order to establish which positions can be measured in a reasonable time interval. This will depend on the relative importance of - epithermal neutron component and - γ dose throughout the assembly.
From these preliminary data, a realistic time schedule to perform the definitive measurements and the physical characteristics of the spectrometer head will be fixed.

Specific problems to be expected

The generally unfavourable experimental conditions of controlled environments and test regions reduce the performances of the differential neutron spectrometry reached in special purpose standard and reference fields.

In the present case the most perturbing parameters are :

- accessibility to the measuring position together with the accompanying perturbation of the neutron field by the spectrometer
- position sensitivity of the neutron spectrum
- angular neutron distribution
- γ-field
- epithermal neutron component.

Solutions to reduce the drawbacks

- The perturbation of the neutron field will be limited by fill-up plugs
- To reduce and/or evaluate the dependency of the spectrometer to the spectral variation either a very small ^{6}Li deposit can be used, which has the considerable drawback of reducing the detection efficiency, or the confrontation of results obtained in different detection modes i.e. detector plane once parallel to the general gradient in the PV mock-up, once perpendicular
- To avoid dependency of the ^{6}Li results to the angular neutron distributions only 4π measurements will be performed
- If one is not only interested in the spectral shape, but also in the flux variation from one position to another, the γ-dose and its variation from one measuring point to another plays a very important role.
  Indeed the tremendous amount of small pulses generated by the γ's introduces in the electronic system a dead time difficult to determine precisely and moreover very sensitive to the degree of stability of the lower levels of the CFTD used in the fast coincidence circuit. To perform feasible measurements the γ-dose should not exceed a given maximum to be determined by the scoping experiment. Dead time correction can then be determined with the aid of a stable pulsgenerator (method tested in the SCK/CEN iron shells). The pile-up effect due to γ's will also be evaluated.
- In presence of an epithermal neutron component the pile-up on the epithermal events raises the lower energy validity limit. Some systematic studies were performed in the iron shells of SCK/CEN [Cft E] showing the possibility to correct for it, if not too important, by the analysis of a pulser peak recorded in the same experimental conditions. Nevertheless preference is given to avoid this correction by reducing this component with the aid of a ^{10}B cover. A preliminary test measurement has been performed in an iron shell configuration. Further measurements as well as calculations will be made to estimate the spectral perturbation due to this cover.

Equipment

The experimental procedure and equipment that will be used is similar to the one used in the iron shell experiment.

The differences with earlier measurements as described in [1, 2] lie in the performances of the analyzer system i.e. a ND 6660 at SCK/CEN and a comparable data analyzer system of Tennecomp. for the PCA measurements.

Expected achievements

Scoping ^{6}Li measurements are programmed from 18 to 30 September. The description of the in-core instrumentation that will be used is given in the PCA Dosimetry Transmittal Letter of April 21, 1978 from R. Gold to F. Kam. A detailed description of the full set-up, together with preliminary observations will be reported after the measurements.

References

[1] EUR 5667 e/f Part 1, p 176-208 (1977).

[2] BLG 501 - G. and S. DE LEEUW.

B. PCA Pressure Vessel Mock-up Characterization :

4. Calculations

G. MINSART

Objective

Support the final design of the facility and the preparation of measurements planned in PCA by computing spatial and spectral neutron and gamma flux distributions in the assembly for different geometries and using several levels of modelling (1-Dim and 2-Dim transport theory).

Summary

The pressure vessel wall mock-up facility in PCA has been computed using a 40-group neutron cross-section library in one-dimensional transport S8 approximation and, after collapsing of cross-sections into a 16-group structure, in one dimension S16 and two dimensions S8 transport approximations.
Neutron flux profiles and spectra at selected positions were analyzed, and spectral indices obtained for a comprehensive set of threshold detectors.
Gamma fluxes were computed for the initially proposed (nominal) configuration and with the Fe thermal shield replaced by a thicker Pb slab.

Parametric 1-Dim studies were performed for different distances between core and thermal shield, and between thermal shield and pressure vessel simulator; the influence of the axial (vertical) buckling used in the model has been also investigated.

The results of these computations have been reported in several notes and letters (see references).

References

[1] "Specification of a 2D x y transport theory model for PCA and ORR-PSF simulated pressure wall mockups". Letter from A. FABRY to G. MINSART, dated July 15, 1977.
"Revised specification ... " Letter dated August 26, 1977.

[2] "Tentative appraisal of your two-dimensional discrete ordinate transport theory calculation of PCA". Letter from A. FABRY to G. MINSART, dated Feb. 9, 1978.

[3] "Two-dimensional calculation of PCA LWR vessel wall experiment". G. MINSART, June 16, 1978.

[4] "Request for two one-dimensional calculations of PCA (modified configurations)". Letter from A. FABRY to G. MINSART, dated May 5, 1978.

[5] Letter from G. MINSART to A. FABRY, dated May 17, 1978.

[6] Letter from G. MINSART to A. FABRY, dated May 22, 1978.

[7] "Theoretical investigations on gamma radiation fields in PCA LWR vessel wall experiment". G. MINSART, August 3, 1978.

C. Iron Shell Benchmark Fields at BR1

G. DE LEEUW-GIERTS, S. DE LEEUW

Objective

The objective of the SCK/CEN iron shell experiments is to provide a reference neutron field for the LWR-PVS program.

Summary

Iron shell reference neutron fields were built at SCK/CEN for the LWR-PV program. They are described in [1] . These neutron fields present the same geometrical and experimental advantages as Mol-ΣΣ . The experimental arrangement is such that the neutron spectrum and nuclear conditions are similar to those met in LWR-PVC experiments and that reactor codes can be applied as such. Nevertheless, because of the simplicity of the set-up, the conclusions drawn from the comparison between theory and experiment are independent of the computational difficulties associated with complex reactor geometries and heterogeneities. The experiment consequently provides a powerful check on adequacy and internal consistency of iron data as used in the reactor computation and on the codes themselves.
The experimental and theoretical analysis is done in collaboration with the NBS.

Status and accomplishment

Description

The reference neutron fields consist of spherical iron shells of variable thickness, driven by a fast neutron spectrum generated by a natural uranium shell surrounding the iron shells. The complete configuration is suspended in a 1 m cavity hollowed in the vertical thermal graphite column of BR1 (Fig. 1).

The uranium shell has 1 cm thickness, the iron shells respectively 7, 9 and 14 cm.

Neutron spectra have been calculated in the center of these 4 configurations by means of the one-dimensional DTF IV code, Kedak 2 library. The initial calculation used a 40 gr set (including twelve groups below 0,95 eV) and assumed a thermal source at the first graphite mesh point. The fission source profile from this calculation was used as input to a second calculation using a 208 g cross section set (Kedak II and Kedak III evaluation). The calculation used 94 mesh points (Fig. 2).

Measurements

The experiment consists in the measurement of the neutron spectrum and reaction rates in the center of each configuration and its comparison with the theoretical spectra computed by means of different reactor codes.

The adequacy of different cross section sets, transformed to group cross sections for spectrum computation is tested by comparing the experimental results to those computed with e.g. Kedak and ENDF/B libraries.

- Two differential neutron spectrometry methods are used to determine the neutron spectrum :
  1. the (n,p) method using small spherical proportional counters with H_2 and CH_4 fillings
  2. the $^6Li(n,\alpha)t$ method using a sandwich of surface barrier detectors.

  To be independent on eventual systematic errors in the spectrometry method e.g. cross section uncertainties, the final analysis will be based on the ratio of the spectra determined in the center of the configurations.

  The spectrum was measured up to now in the 1 cm U_{nat} + 7 cm Fe by means of both methods. Preliminary measurements were performed in the 1 cm U_{nat} and 1 cm U_{nat} + 14 cm Fe (here only Li) configurations.

- Fission reaction rates have very recently been performed by an NBS expert in these configurations.
  Fission rate distribution will also be measured within the shells by means of SSTR's.

SCK/CEN iron shells and PCA spectral comparisons

The neutron spectra in PCA and in the iron shells are very similar. For illustration table 1 compares the mean cross sections of the $^{6}Li(n,\alpha)t$, ^{5}U and ^{8}U fission reactions computed for - the 1 cm U_{nat} driver, 1 cm U_{nat} + 7 cm Fe and 1 cm U_{nat} + 14 cm Fe configurations and for 3 PCA positions - front of PV mock-up, 1/4 and 1/2 positions [2].

$^{6}Li(n,\alpha)t$ cumulative responses are also given in large energy bins so as to be able to compare the general shape of the neutron spectra. Fig. 3 compares the spectra in the center of the 1 cm U_{nat} driver to the spectrum incident in the PV mock-up. Because of the similarity, the problems encountered in the iron shell measurements will also exist in the PCA experiment and the solutions adopted, directly applicable to the PCA data.

Preliminary to the PCA experiment (cft. G.2), where the epithermal and γ component are probably more important, a method for pile-up and dead time corrections was tested in several U and Fe configurations and at several reactor powers.

A ^{10}B cover was manufactured and was used, for testing, in the 1 cm U_{nat} + 7 cm Fe configuration. A reduction of the epithermal ^{6}Li reaction peak by a factor ~ 5 was observed relative to the measurements with a Cd cover. More measurements will be performed to increase statistics in the fast neutron part of the spectrum.

Expected achievements

To be able to determine the spectrum by means of the (n,p) method in the 1 cm U_{nat} + 14 cm Fe configuration, smaller spherical proportional counters have been ordered. The (n,p) measurement will be restarted as soon as these counters arrive.

- A new data analyzer system much more powerful than the one presently in use was purchased. It will allow to improve the ^{6}Li data and to determine, because of the increased core memory, the neutron spectrum in more narrow energy interval below 1 MeV.

  The Li-6 measurements will restart as soon as the system is operational.

- Relative ^{5}U fission rate measurements will be performed in all configurations to determine precisely the Cd ratio's.

References

[1] G. De Leeuw-Gierts, S. De Leeuw
IAEA-208, pp. 375-383

[2] G. Minsart, private communication .

TABLE 1

	Average sigma (total flux = 1)					
	1 cm U_{nat}	PCA incident PV	1 cm U_{nat} + 7 cm Fe	1/4 thickness	1 cm U_{nat} + 14 cm Fe	1/2 thickness
$^{235}U(n,f)$	2.48 +2	5.39 +2	1.54 +1	2.96 +1	4.94	7.43
$^{238}U(n,f)$	7.38 -2	3.95 -2	7.13 -2	7.34 -2	3.88 -2	4.82 -2
$^{6}Li(n,\alpha)t$	4.06 +2	8.47 +2	2.37 +1	4.42 +1	5.71	8.7
$E_{n_{low}}$ (MeV)	Cumulative $^{6}Li(n,\alpha)t$ responses					
1.0 -9	1	1	1	1	1	1
9.5 -7	5.88 -3	6.63 -3	1.32 -1	1.55 -1	5.91 -1	5.75 -1
1.0 -5	2.53 -3	1.86 -3	8.22 -2	6.86 -2	3.45 -1	2.96 -1
1.0 -3	8.74 -4	3.03 -4	3.35 -2	1.77 -2	1.6 -1	1.04 -1
1.0 -2	6.86 -4	1.66 -4	2.85 -2	1.34 -2	1.42 -1	8.53 -2
[illegible] 1	1.83 -4	4.42 -5	3.47 -3	1.74 -3	9.15 -3	6.5 -3

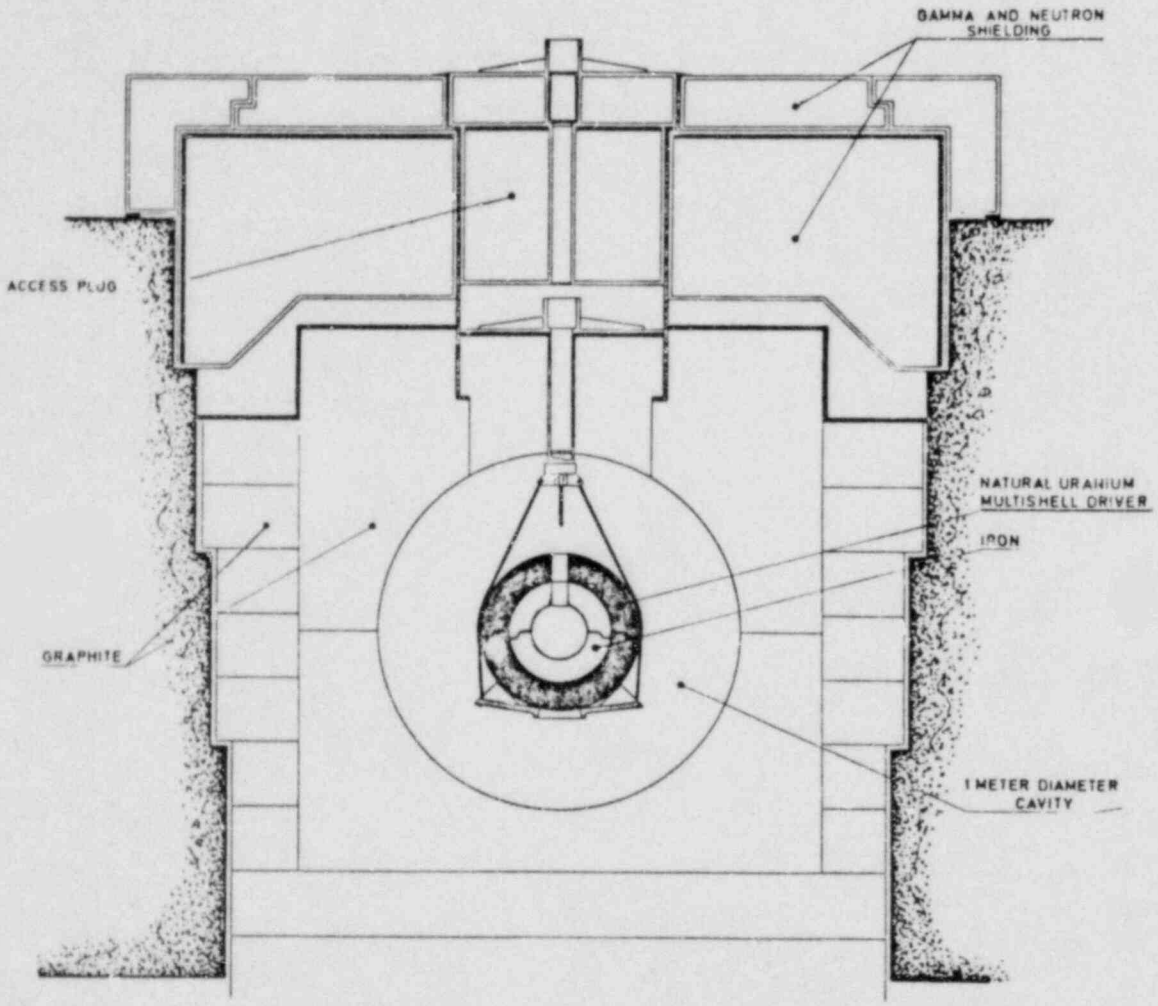

GAMMA AND NEUTRON SHIELDING
ACCESS PLUG
NATURAL URANIUM MULTISHELL DRIVER
IRON
GRAPHITE
1 METER DIAMETER CAVITY

Fig. 1

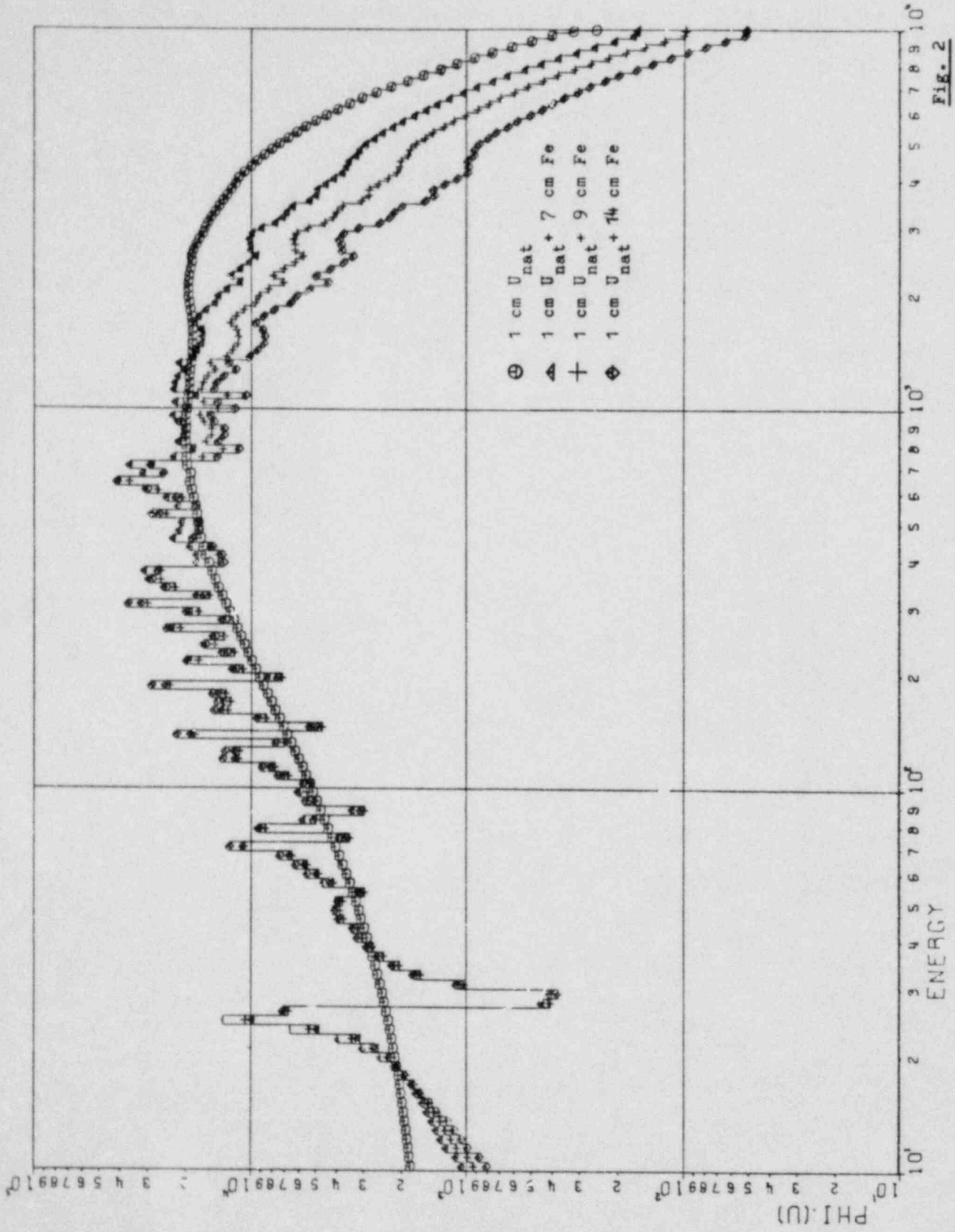
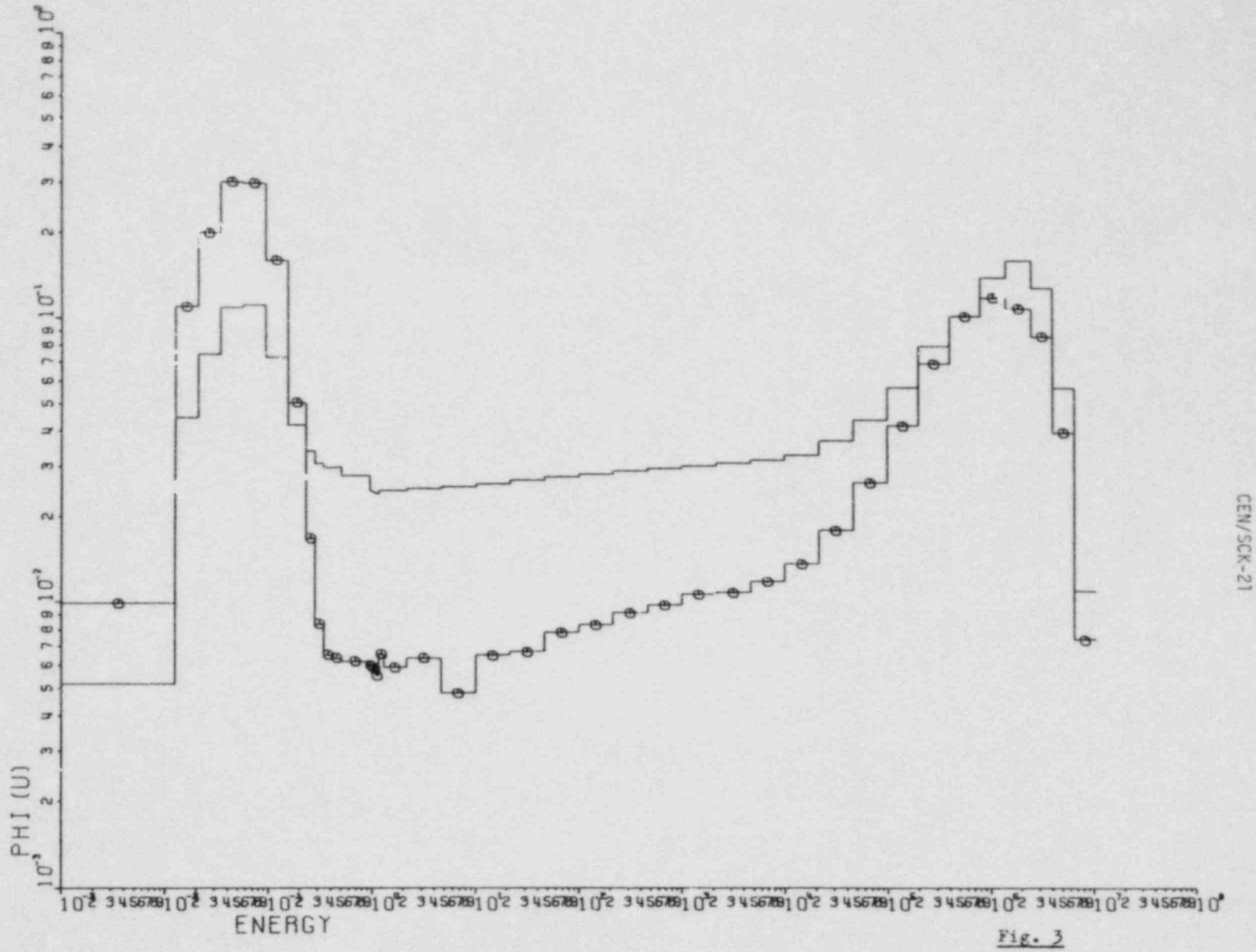

1 cm U_{nat}
1 cm U_{nat} + 7 cm Fe
1 cm U_{nat} + 9 cm Fe
1 cm U_{nat} + 14 cm Fe
PHI.(U)
ENERGY

Fig. 2



CEN/SCK-21

Fig. 3

COMPARISON FOR INCIDENT ON PRESSURE VESSEL WITH 1 CM DRIVER FROM16.25

D. Irradiations in FRJ-1 and FRJ-2 (Jülich)

H. TOURWE, A. FABRY

Objectives

- Establish the neutron spectral sensitivity of irradiation embrittlement for a few relevant LWR pressure vessel steels which have also been selected for exposure in the ORR-PSF pressure vessel wall mock-up.
- Make a tie of the IAEA coordinated program with the NRC program and the BR3 program.
- To appreciate the consistency of dosimetry techniques used in LWR environmental conditions through interlaboratory comparison.

Summary

It was decided during recent discussions at Jülich (April 24-26, 1978) between KFA, HEDL and SCK/CEN representatives that joint pressure vessel steel irradiations with an interlaboratory dosimetry should take place in the FRJ-2 (D_2O) and the FRJ-1 (H_2O) reactors at Jülich. The fluence target (> 1 MeV) for these irradiation is $2.10^{19} n/cm^2$. Three different irradiation locations are available :

1) hollow fuel element in FRJ-2 core (~ 10 d exposure)
2) reflector FRJ-2 (~ 100 d exposure)
3) hollow fuel element in FRJ-1 core (~ 10 d exposure).

Two rigs with steel specimens and dosimeters will be loaded at each location in FRJ-2. One rig is foreseen for FRJ-1.

Timing schedule

The two rigs in FRJ-2 core will be irradiated in October 1978.
The FRJ-2 reflector irradiation of the first rig will start in October 1978; the irradiation of the second rig in February 1979. The exposure in the FRJ-1 core is planned for July 1979.

It is planned to perform at each of the 3 locations a short time irradiation at low power for neutron spectrum measurements just before the start of the corresponding long-time irradiations.

Accomplishments and status

FRJ-2 rig description

In each rig is space for 20 charpy's and 4 tensiles. There are 4 layers with steel specimens and each layer can accommodate 5 charpy's and one tensile (fig. 1). Two layers are situated above the core midplane; two layers are situated below the core midplane.

Three different types of steel will be irradiated in these rigs :

1) SA533B cl. 1 plate HSST-03 : this steel is part of the KFA IAEA involvement and is already available at KFA.
2) SA508 cl. 3 forging : this steel is supplied by the SCK/CEN. The specimens were prepared by Cockerill-Ougrée in collaboration with the Centre de la Recherche Métallurgique (Liège). Unirradiated data on this type of steel are given elsewhere in the SCK/CEN contribution to the "LWR Pressure Vessel Irradiation Surveillance Dosimetry" progress report.
3) SA302B plate : this relatively high sensitive steel is provided by NRC.

The number of charpy's and tensiles from the different types of steel loaded into the FRJ-2 rigs are given in table I. An interlaboratory dosimetry capsule will occupy the space of one tensile in every rig. All tensiles are fabricated as indicated in the DIN 50125 standard. The tolerances for the charpy specimens are given in fig. 2.

The rigs will be rotated by 180° around their vertical axis as soon as the half irradiation time has been reached in order to eliminate the effect of the radial flux gradient.
The rigs are positioned during the irradiation in a temperature controlled environment of (290 ± 10)°C.

FRJ-1 rig description

The lay out of this rig has to be discussed in more detail. Up to now it was decided that the same types of steel will be loaded into this rig, while the position of one tensile specimen will be replaced by an interlaboratory dosimetry capsule. The number of metallurgic specimens loaded into this capsule will probably be about two times higher than in one FRJ-2 rig.

Dosimetry

It is the responsibility of KFA to perform at each irradiation location a short time irradiation at low power for neutron spectrum measurements. The rig is inserted in the reactor during this irradiation while 4 of the charpy specimens are replaced by hollow charpy specimens with activation detectors. The used activation reactions for this neutron spectrum measurement are given in table II.
During the long term irradiation an interlaboratory dosimetry capsule, with the same outer dimensions as a tensile specimen, takes the place of a tensile specimen. The design of this capsule is shown in fig. 3 : the HEDL-AI, the KFA and the MOL dosimetry sets are inserted in a gadolinium filter, while this filter on his turn is placed in a stainless steel holder. Gradient foils are placed between the different sets for axial gradient measurements. The final assembling and the welding of the interlaboratory capsules is done at Jülich. The welding is performed under argon atmosphere. The HEDL dosimeters and the AI helium accumulation samples were sent to Mol for assembling. There were mounted in an aluminium box, wrapped in a thin aluminium foil (see fig. 4 a). The HEDL wires were wrapped in a thin aluminium foil before mounting them into the boxes in order to facilitate the identification between HEDL and AI wires after the irradiation. The types of dosimeters loaded into the HEDL-AI sets are summarized in table III.

The design of the holder of the MOL dosimeter set is shown in fig. 4 b. The maximum temperature during irradiation is measured with a set of 4 melting wires :

Bi	: melting point (271 ± 3)°C
93 % Cd - 7 % Sb	: melting point (290 ± 5)°C
97,5 % Pb - 1,75 % Ag - 0,75 % Sn	: melting point (310 ± 3)°C
Pb	: melting point (327 ± 3)°C

So the maximum temperature can be measured with an accuracy of 10°C. The types of dosimeters loaded into the HEDL-AI sets are summarized in table IV. The KFA dosimeter sets will contain no ^{5}U, ^{8}U and ^{7}Np dosimeters. The ^{5}U, ^{8}U and ^{7}Np dosimeters loaded in the Mol sets will after irradiation be measured by Jülich and Mol. All details about the KFA dosimeter sets are not yet available.
Interlaboratory gradient sets will be inserted between the first and the second layer and between the third and fourth layer of steel specimens during the FRJ-2 irradiations. Six sets are loaded at each level (each set is sandwiched between the upper and lower steel specimen) : one MOL set, one HEDL set and four KFA sets. The MOL and HEDL set will occupy opposite positions on the axis "core centre" - "rig centre".
1 % Co/Al, 1 % Ag/Al and Fe foils from the MOL-dosimeter stock are loaded in the MOL and HEDL gradient sets. Mol has supplied the necessary amount of dosimeters to KFA. KFA is responsible for the preparation and the assembling of all gradient sets.

Expected achievements

Mol will supply detailed information on the MOL and HEDL dosimeter sets to all the involved laboratories. Some details about the FRJ-2 and FRJ-1 irradiation will be discussed with KFA-people. KFA will perform the final assembling for the FRJ-2 irradiation and will send a detailed loading scheme to the involved laboratories.

Table I : number of charpy's and tensiles in the different FRJ-2 rigs

position / steel type	FRJ-2 core				FRJ-2 reflector			
	rig 1		rig 2		rig 1		rig 2	
	charpy's	tensiles	charpy's	tensiles	charpy's	tensiles	charpy's	tensiles
SA533B cl 1 (plate HSST-03)	13	2			13	2		
SA508 cl 3 (forging)	7	1	7	1	7	1	7	1
SA302B plate			13	2			13	2
TOTAL	20	3	20	3	20	3	20	3

TABLE II : activation reactions used for neutron spectrum measurements in the low power runs

Sc^{45} (n,γ)Sc^{46}

Au^{197}(n,γ)Au^{198}

Co^{59} (n,γ)Co^{60}

Fe^{58} (n,γ)Fe^{59}

Cu^{63} (n,γ)Cu^{64}

Pu^{239}(n,f)

U^{235} (n,f)

Np^{237}(n,f)

In^{115}(n,n')In^{115m}

Th^{232}(n,f)

U^{238} (n,f)

Ti^{47} (n,p)Sc^{47}

Ni^{58} (n,p)Co^{58}

Fe^{54} (n,p)Mn^{54}

Ti^{46} (n,p)Sc^{46}

Fe^{56} (n,p)Mn^{56}

Al^{27} (n,α)Na^{24}

Ti^{48} (n,p)Sc^{48}

Table III : types of dosimeters in the HEDL-AI sets

a) HEDL dosimeters

0.116 % Co/Al wire
0.145 % Ag/Al wire (2)
Ni wire
Ti wire
Cu wire
Fe wire
Sc_2O_3, $^5UO_2$, NpO_2, $^8UO_2$ } encapsulated in V

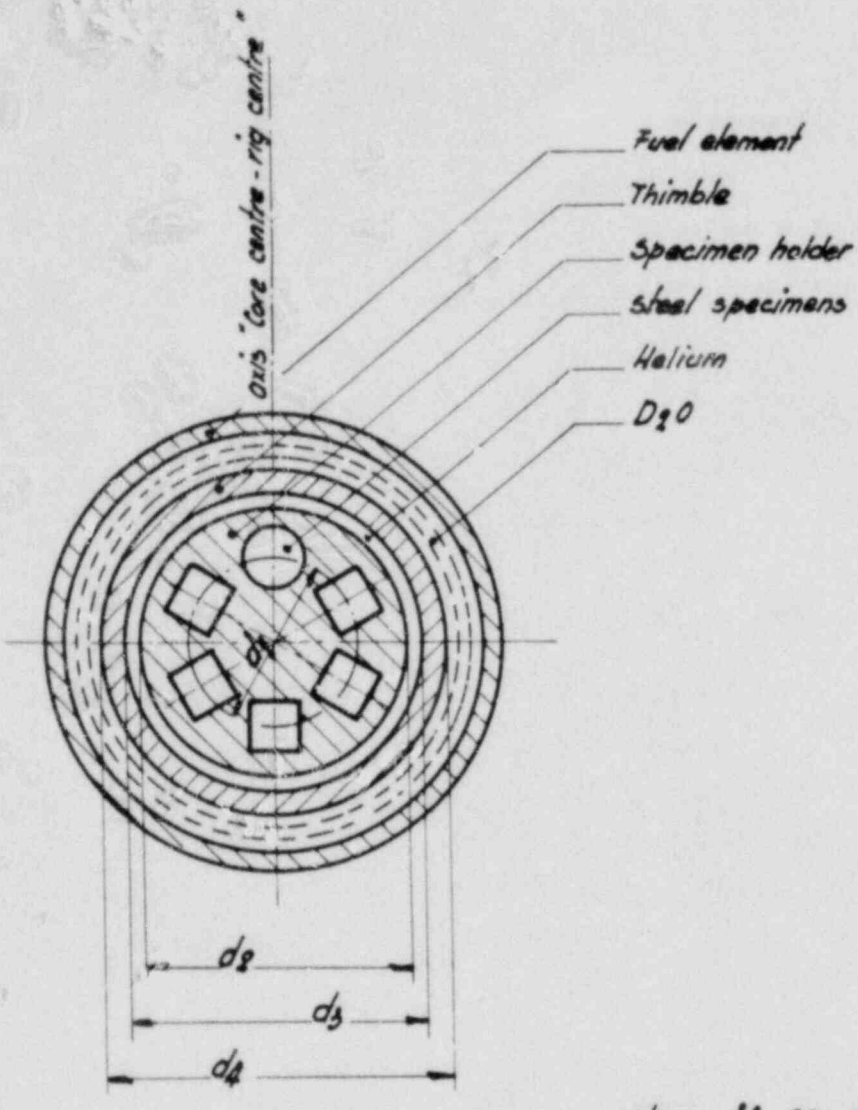
b) AI samples

Sample		FRJ-2 core FRJ-2 reflector rig 1	FRJ-2 core FRJ-2 reflector rig 2	FRJ-1 core
Fe wire		X	X	X
Ni wire		X	X	X
Cu wire		X	X	X
Al wire		X	X	X
0.13 % ^{10}B/Ni wire		X	X	X
0.6 % ^6Li/Al wire		X	X	X
Ti wire		X		X
Nb wire		X		X
Pt wire		X		X
Zr chunks		X		X
0.13 % ^{10}B/Ni	vanadium encapsulated	X		X
0.6 % ^6Li/Al		X		X
blank		X	X	X
TiN		X	X	X
PbS		X	X	X
Be		X (2)	X	X (2)
ZrN (lot A)		X	X	X
Cu_2S		X	X	X
ZrN (lot B)		X	X	X
FeF_2		X (2)		
V_2O_5		X		
CaF_2			X (2)	
KCl			X (2)	

Table IV : types of dosimeters in the MOL-sets

- $^{5}UO_2$, NpO_2, $^{8}UO_2$, Sc_2O_3 } encapsulated in vanadium
- 1 % Co/Al foil
- 1 % Ag/Al foil
- Ni foil
- Fe foil
- Ti foil
- Cu foil
- Quartz
- Nb foil (Max Planck Institute)
- Nb foil (Goodfellows) }
- Nb foil (Highways) } *
- Nb foil (Kawecki) }

*only in FRJ-2 core/rig 1 and FRJ-2 reflector/rig 1



d_1 : 33.00 mm
d_2 : 50.09 mm
d_3 : 50.80 mm
d_4 : 52.00 mm

Fig. 1: Horizontal section of the FR3-2 rig.

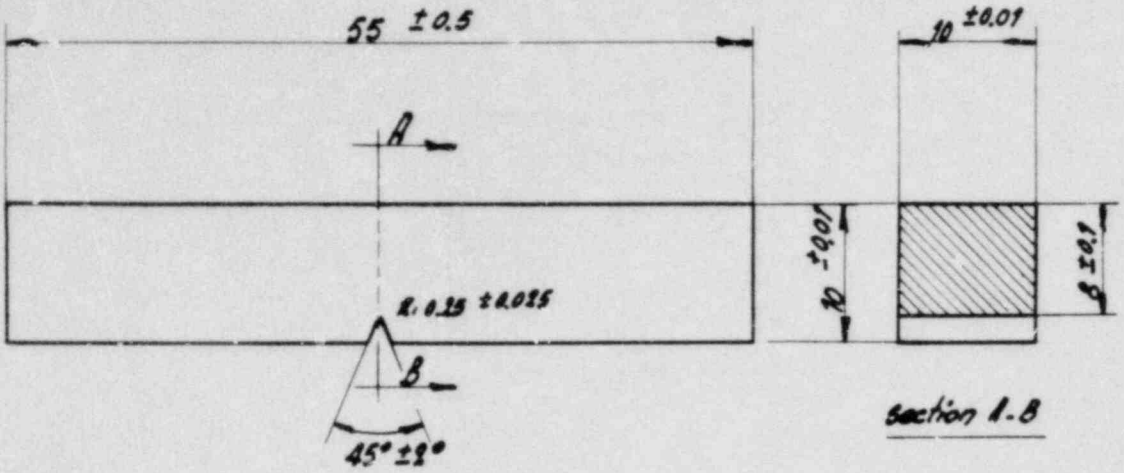


Fig. 2. Dimensions and tolerances of the charpy specimens.

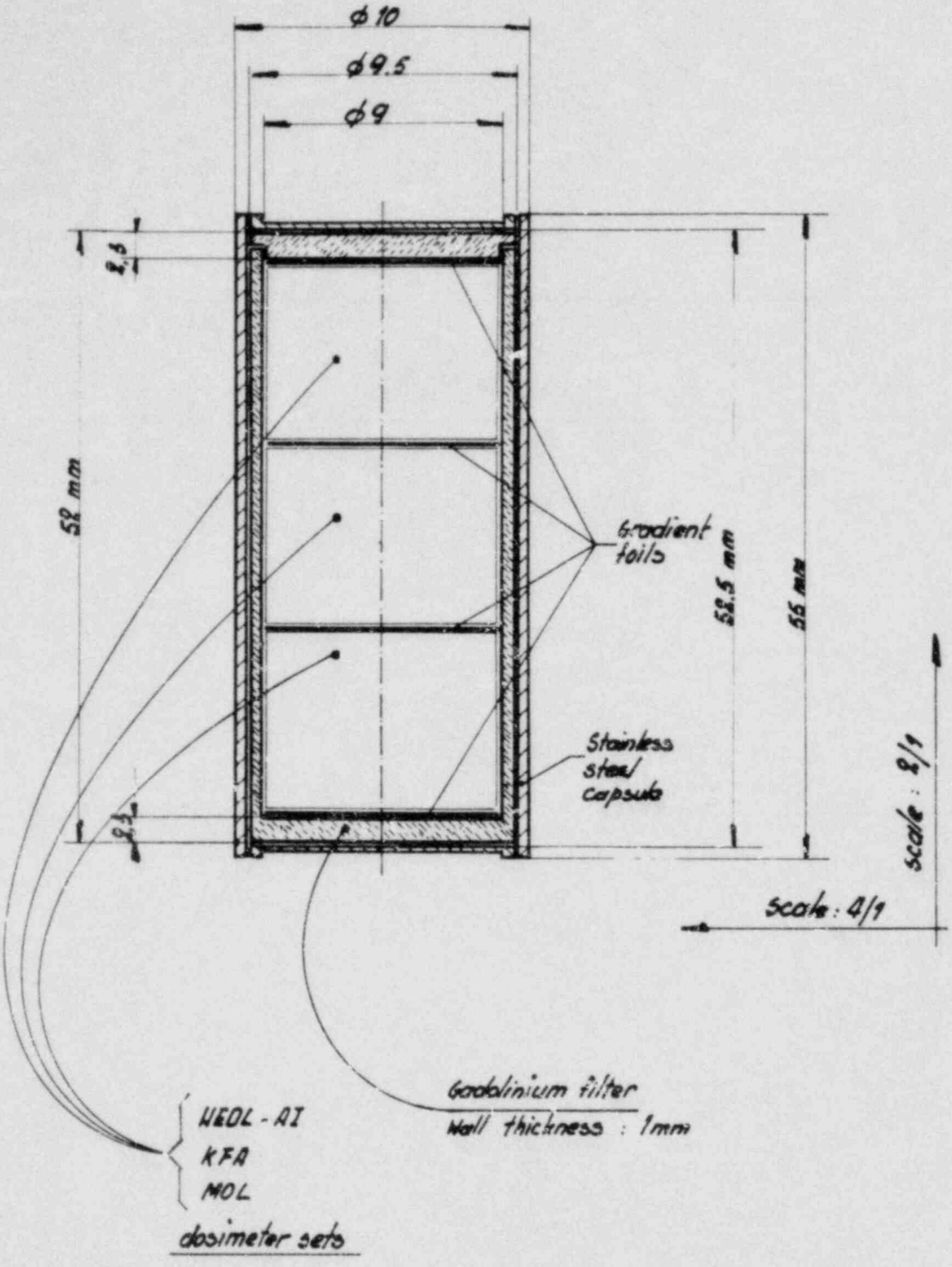


Fig. 3 Design of the interlaboratory dosimetry capsule

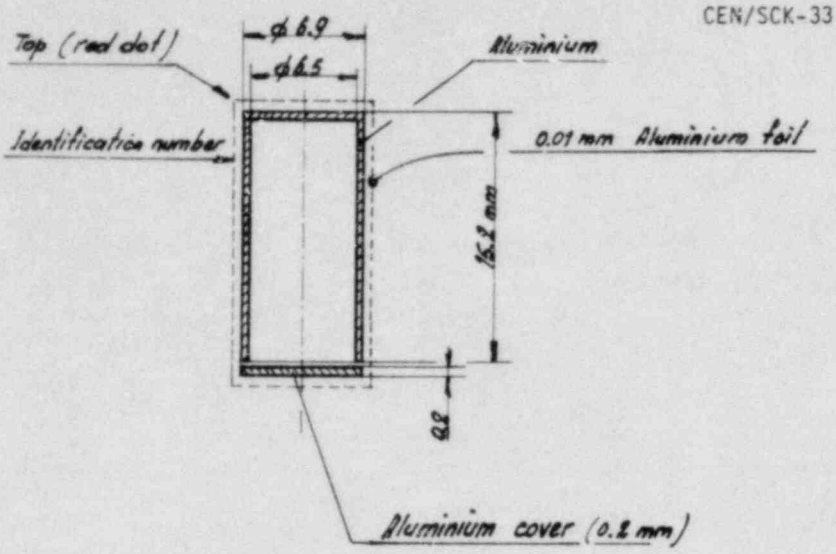


Fig. 4a : HEDL - AI set holder

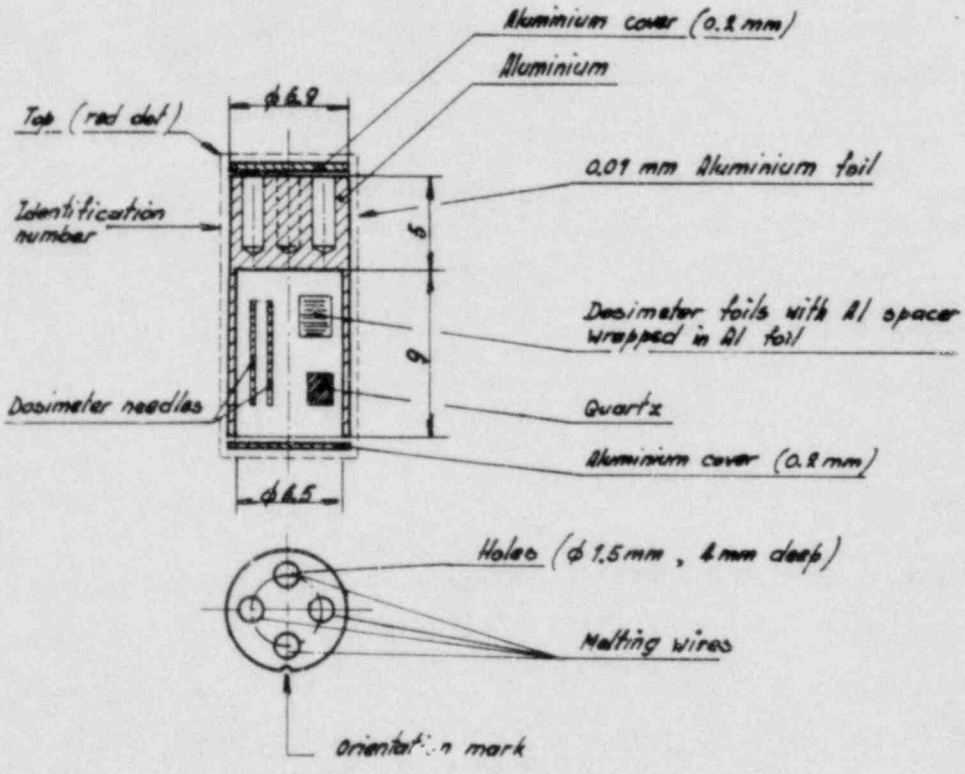


Fig. 4b : MOL set holder

E. BR3 Test Regions Program

H. TOURWE, J. DEBRUE, P. GUBEL, A. FABRY, J. MINSART, Ph. VAN ASBROECK
Technical assistance : P. VANMECHELEN

Objectives

The objectives of the dosimetry programme and of the irradiations of RPV steel specimens in BR3 core 4B are :

- to provide neutron fluence data at selected positions inside and outside the core in order to validate or improve calculation methods used in PWR surveillance applications
- to appreciate the consistency of dosimetry techniques used in PWR environmental conditions through interlaboratory comparisons
- to provide irradiation effect data for a specific steel (A508, class 3) and verify the validity of the testing procedure as a whole (including temperature monitoring, mechanical tests, dosimetry), with the aid of the simultaneous irradiation of a HSST reference steel (A533 grade B, from HSST plate 02)
- to have a better knowledge of the BR3 vessel exposure level.

Summary

The BR3 core 4B will be operated during about 11 months in the period 78-79. The steel specimens and dosimeters will be irradiated during the whole period.

A cooperative link has been established with the U.S. Pressure Vessel Irradiation Surveillance Dosimetry program for the dosimetry work and related calculations. From the U.S. side, dosimeters were provided by HEDL, GE and AI and neutron transport calculations will be performed by EPRI.

A HSST block from the A533 grade B plate C2 was made available by ORNL, at the request of the Nuclear Regulatory Commission; charpy V-notch specimens were taken out of this block at the 1/4" thickness of the original plate.

The irradiation of A508 class 3 specimens is of interest for the Belgian industry (Cockerill-Ougrée Providence) which took the responsability of preparing all the specimens, including the HSST ones, in collaboration with the Centre de la Recherche Métallurgique (Liège). The A508 class 3 specimens will also be of the charpy V-notch type (base metal, heat affected zone and welding).

C.E.N./S.C.K. is in charge of the irradiation, of dosimetry and related analytical work, and of the post-irradiation mechanical tests.

Accomplishments and Status

Irradiation locations available for steel samples and dosimeters

BR3 is a pressurized water moderated reactor, operated at a nominal thermal power of 40.9 MW. The average water temperature is 260°C in the core region and the pressure amounts to 140 kg/cm^2.

The general shape of the core configuration is hexagonal; it is made of 91 hexagonal cells (fig. 1 and 2), 73 of which containing fuels. Eight out of the 18 remaining cells, which are called moderator tubes (water filled channels) can be provided with fluence monitor wires located on the axis of the channel, over the entire height of the core. Baskets with steel specimens can moreover be irradiated in the upper part of these moderator tubes (numbered 8, 12, 27, 29, 63, 65, 80, 84 on fig. 2), at the level of the axial reflector; these locations are called the "Low Flux Positions". Four positions (8, 29, 63 and 84) will be used for the specimens of the present programme; 18 charpy specimens with their greater length (55 mm) parallel to the channel axis, will take place at each position over a total length of 110 mm (there are therefore two groups which are at different distances from the core midplane, see fig. 3). The distance of 695 mm was defined with the objective of achieving a neutron dose of about $2 \times 10^{19} n/cm^2$ (> 1 MeV) at the low flux positions.

The High Flux Positions are located in the radial reflector (fig. 2 and 3). Position 4 will be used for irradiating two sets of steel specimens (one set of 12 specimens and one set of 14 specimens) at about the same exposure level but at different temperatures; moreover, dosimeters will be located in the vicinity of the reactor midplane for "unperturbed" fluence measurements. Dosimeters will also be irradiated (see fig. 2) :

- in the thermoelectric couple channel A, close to the core basket
- in the cold-leg position 1 and 2 between the core basket and the thermal shield*
- in the position F between the pressure vessel and the neutron shield tank*
- in the position I (120°) outside the neutron shield tank.

Loading scheme of the charpy V-notch samples

Low flux (LF) positions

The major objective is to obtain during the cycle BR3/core 4B a fast neutron dose (> 1 MeV) on the charpy's of $\sim 2.10^{19} n/cm^2$. The midpoint of these capsules is situated 695 mm above the core midplane. The loading scheme of the charpy's with the expected neutron flux and neutron fluence values is given in table I. Each capsule has an overall length of 110 mm and consists of two layers of specimens (9 samples in every layer). The arrangement of the samples in each layer and their orientation with respect to the core axis is given in fig. 4a for the low flux positions 29 and 63 and in fig. 4b for the low flux positions 8 and 84. Fig. 5 shows an overall view of the orientation of the specimens with respect to the core axis.

The maximum temperature inside each capsule is measured with melting wires. One set of melting wires is loaded per layer of specimens, i.e. two sets per capsule. One set of melting wires consists of the following materials :

---

*At the time of writing, the possibility of inserting dosimeters in these positions is only tentative. Real testing "in situ" will only be possible just after core loading.

Bi	: melting point	(271 ± 3)°C
93 % Cd - 7 % Sb	: melting point	(290 ± 5)°C
97.5 % Pb - 1.75 % Ag - 0.75 % Sn	: melting point	(310 ± 3)°C
Pb	: melting point	(327 ± 3)°C

The maximum temperature during irradiation can be determined with an accuracy of ± 10 °C.

High flux (HF) positions

Two capsules, one containing 12 steel samples and one containing 14 steel samples, are loaded into the HF position 4. Each capsule has an overall length of 110 mm and consists of two layers of specimens (respectively 6 and 7 samples in every layer).
The loading scheme of the charpy's, the expected neutron flux and neutron fluence values at the midpoint level of each capsule are given in table I. The expected flux gradient between the midpoints of the charpy's in the two layers of the same capsule is about 30 %. The arrangement of the samples in each layer and their orientation with respect to the core axis are given in fig. 6a and fig. 6b.
The temperature measurements in the HF capsules are performed in the same way as for the LF capsules.

Loading scheme of the dosimetry capsules

Neutron dosimetry intercomparisons, involving MOL, HEDL, AI and GE, will be performed at four selected locations :

- LF positions 29 and 63 (level + 695 mm)
- in the radial reflector, position HF 4 (level + 470 mm)
- outside the neutron shield tank, position I (120°).

The orientation of the HEDL-GE and HEDL-AI capsules with respect to the MOL capsules and with respect to the core axis is shown in the figures 4a, 6a and 7. The HEDL-AI sets are loaded on top of the HEDL-GE sets in the LF and HF positions.

Only Mol dosimeter sets are loaded at all the other locations. There are in total five different Mol dosimetry sets involved :

- set S1 : shown in fig. 8
- set S2 : shown in fig. 9
- set S3 : shown in fig. 7
- set S4 : shown in fig. 7
- set S5 : this is a small SS tube Ø 1.1/0.8 containing the following needles : Fe, Ni, Nb, Cu, Al/Co 1 %, Al/Ag 1 %, Fe .

The total loading scheme of the different MOL dosimeter sets is given in table II. Besides these dosimeter sets, one Fe wire (~ 3m) and one 1 % Co/Al wire (~ 3 m) will be placed in position F along the pressure vessel wall.

Neutronic calculations

The geometrical model of the BR3 reactor during BR3 core/4B was prepared. This geometrical model and the neutron calculations procedures were discussed with EPRI. It was decided that calculations will be performed by SCK/CEN Mol and EPRI on 1/6 of the reactor core in Rθ geometry. These calculations will be performed for the start of life and the end of life conditions.

Expected achievements

The start-up of the BR3 core/4B is actually foreseen for late this fall. The first experimental results can be expected during the first quarter of 1981. It is expected that the data concerning the core composition (type of fuel element and burn-up) will be made available within the next 6 months.

TABLE I : LOADING SCHEME OF THE SPECIMENS IN THE BR3 REACTOR

Position	Coolant Temperature (°C)	Samples *		Neutron flux > 1 MeV ($n\ s^{-1}\ cm^{-2}$)	Integrated neutron flux > 1 MeV ($n\ cm^{-2}$)	Capsule midpoint level (mm)
		HSST	508 class 3, HAZ and weldings			
LOW FLUX POSITIONS						
LF 8	~ 260	3X 3X	3B + 3Z 3B + 3Z	$5,7\ 10^{11}$	$1,6\ 10^{19}$	+ 695
LF 84	~ 260	3X 3X	3B + 3Z 3B + 3Z	$5,7\ 10^{11}$	$1,6\ 10^{19}$	+ 695
LF 29	~ 260	3X 3X	3B + 3W 3B + 3W	$6,9\ 10^{11}$	$2,0\ 10^{19}$	+ 695
LF 63	~ 260	3X 3X	3B + 3W 3B + 3W	$6,9\ 10^{11}$	$2,0\ 10^{19}$	+ 695
HIGH FLUX POSITIONS						
Position	Coolant Temperature (°C)	Samples *		Neutron flux > 1 MeV ($n\ s^{-1}\ cm^{-2}$)	Integrated neutron flux > 1 MeV ($n\ cm^{-2}$)	Capsule midpoint level (mm)
		HSST	508 class 3, HAZ and weldings			
HF 4	~ 270	4X	8B	$2,4\ 10^{12}$	$7,0\ 10^{19}$	+ 470
HF 4	~ 280	6X	8B	$2,4\ 10^{12}$	$7,0\ 10^{19}$	- 470

* Meaning of the symbols X, B, Z and W :

- X HSST 02
- B 508 class 3, base metal
- Z 508 class 3, HAZ
- W weldings

TABLE II : LOADING SCHEME OF THE MOL DOSIMETER SETS IN THE BR3/CORE 4B

Level with respect to core midplane (mm)	CORE						RADIAL REFLECTOR		Cold leg Pos. 2	Cold leg Pos. 1	Pos. I (120°)
	Cell 8	Cell 84	Cell 29	Cell 63	Cell 27	Cell 65	Pos. 4	Pos. A			
- 500									S2		
- 470 (HF)							2 x S2				
- 250									S2		
- 235							S2				
- 28											2 x S4
0	S2	S2	S2	S2	S2	S2	S1		S1	S2	2 x S3*
+ 25								S5			
+ 28											2 x S4
+ 180				S2							
+ 235							S2				
+ 250									S2		
+ 360				S2							
+ 470 (HF)							2 x S1*				
+ 500									S2		
+ 695 (LF)	2 x S2	2 x S2	S1 S2 *	S1 S2 *							
+ 842					S2	S2					

*positions where interlaboratory comparison is performed

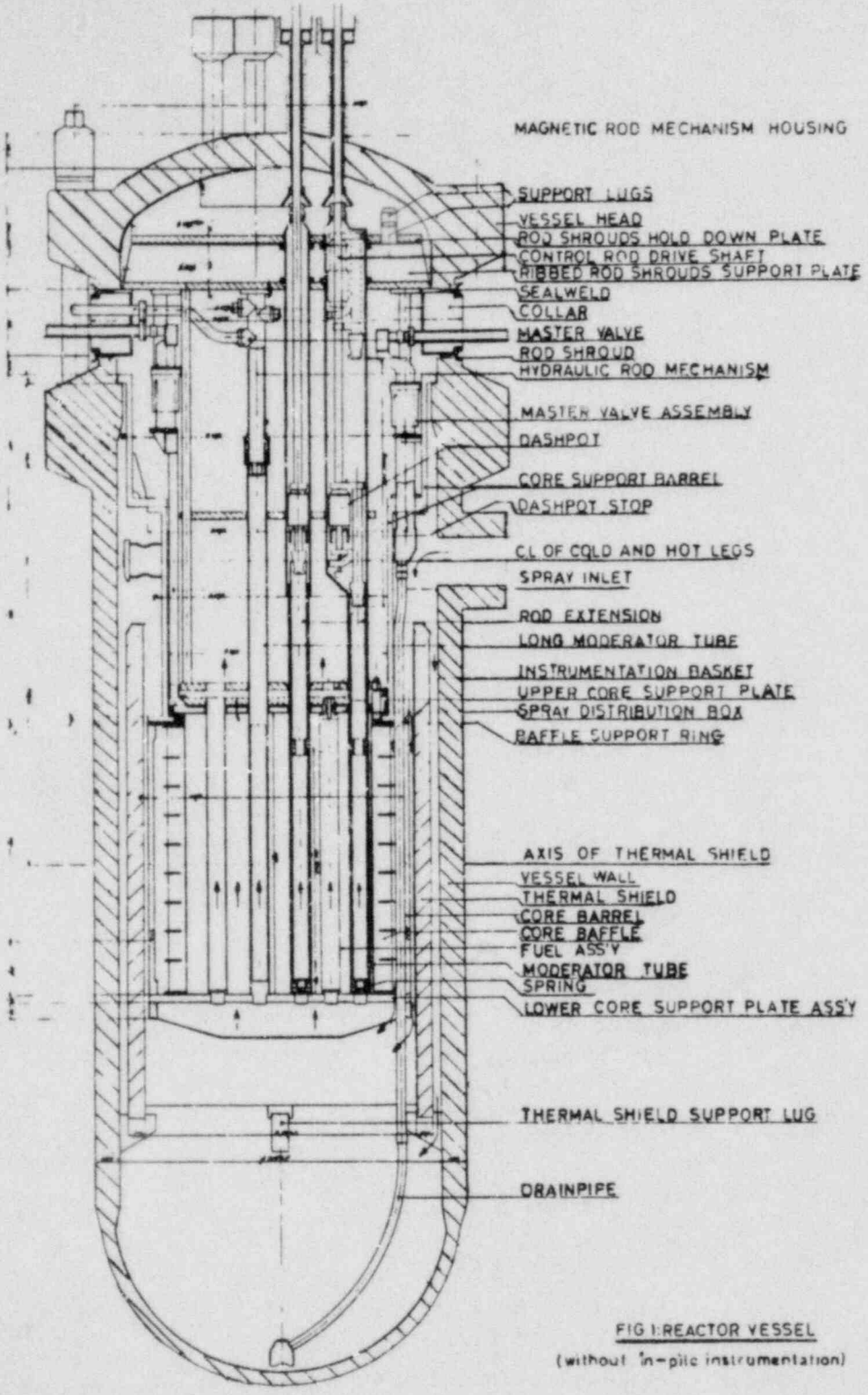


FIG 1: REACTOR VESSEL
(without in-pile instrumentation)

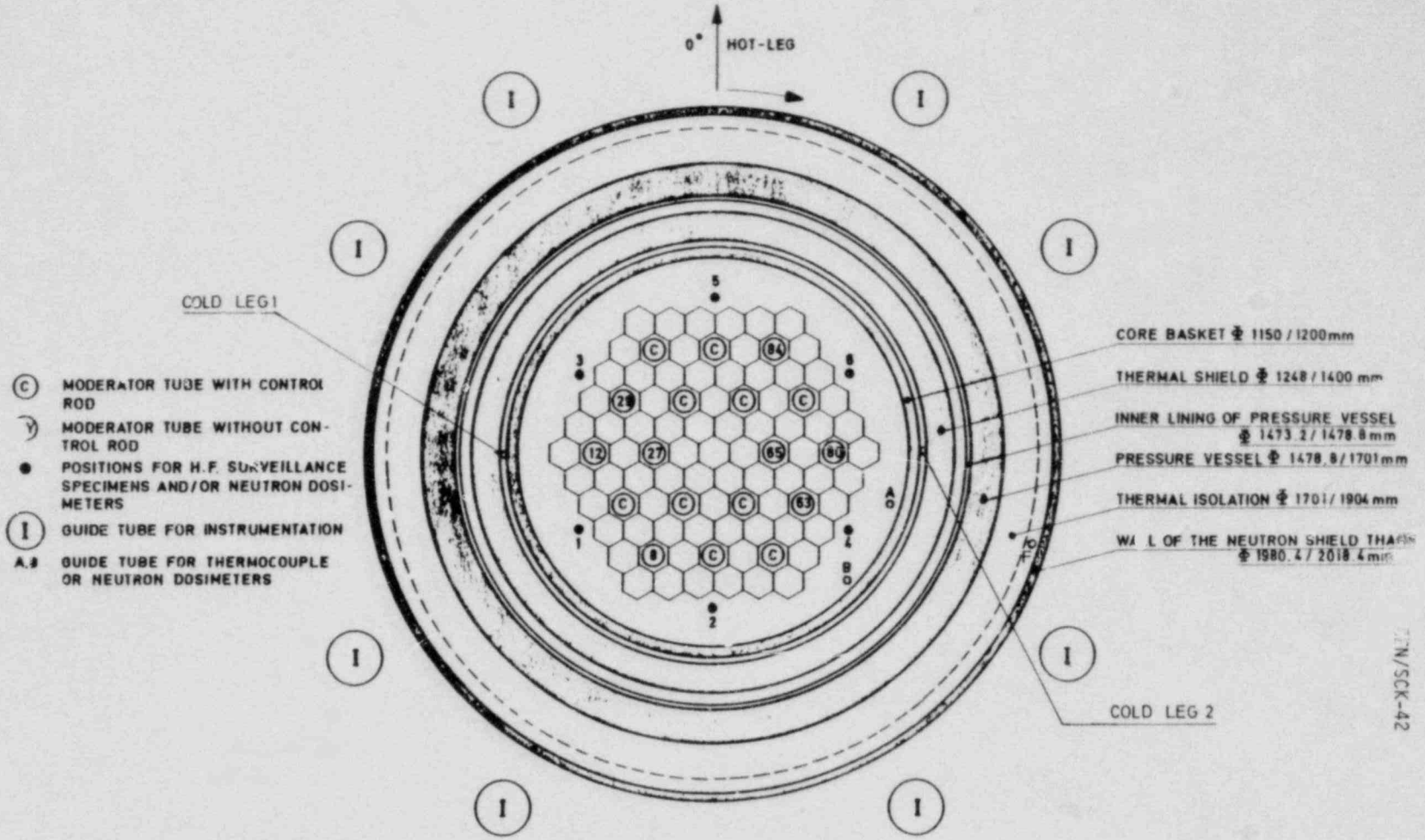


FIG. 2 : BR3 CORE

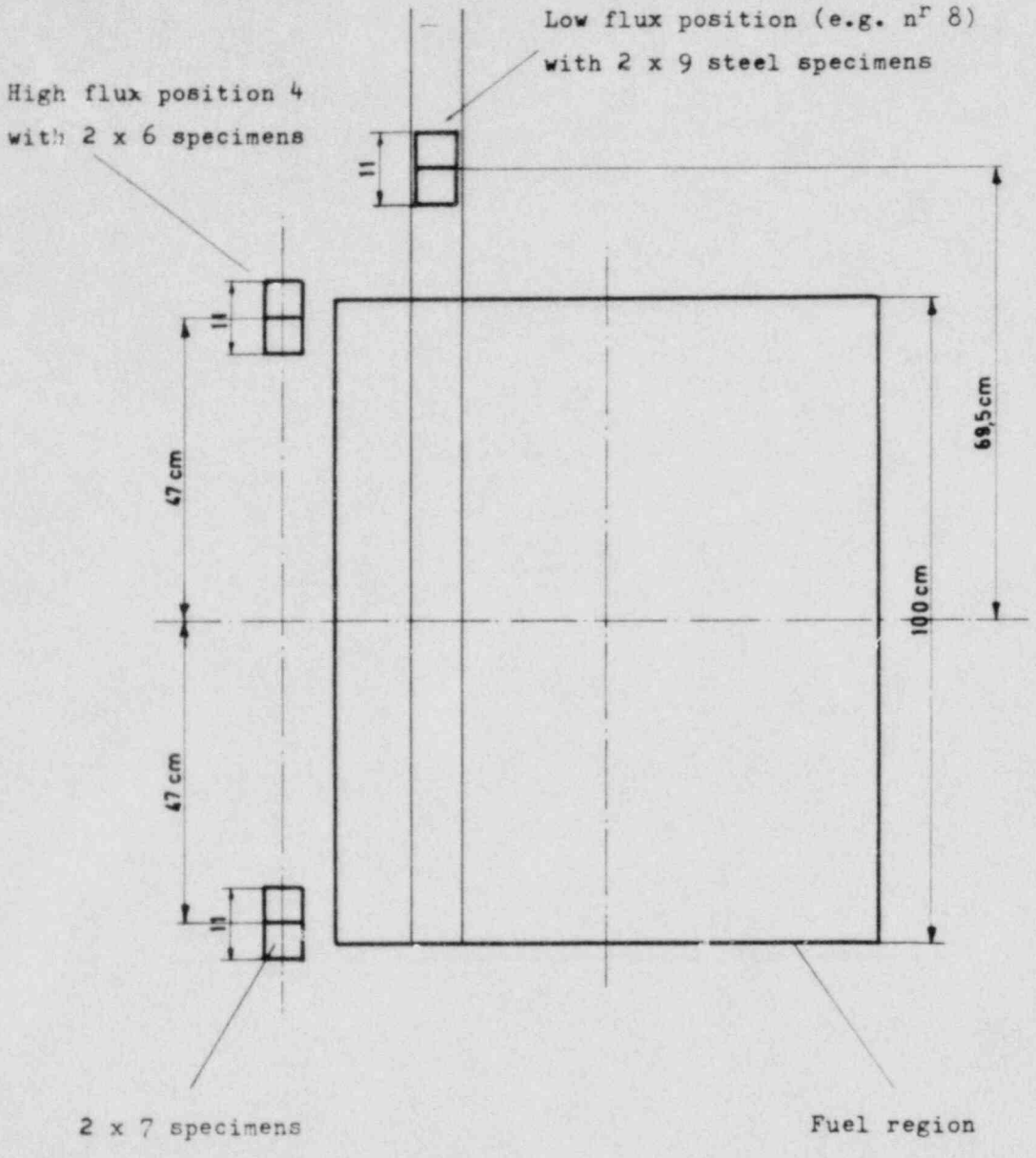


Fig. 3 : Schematic representation of low and high flux irradiation positions for steel specimens

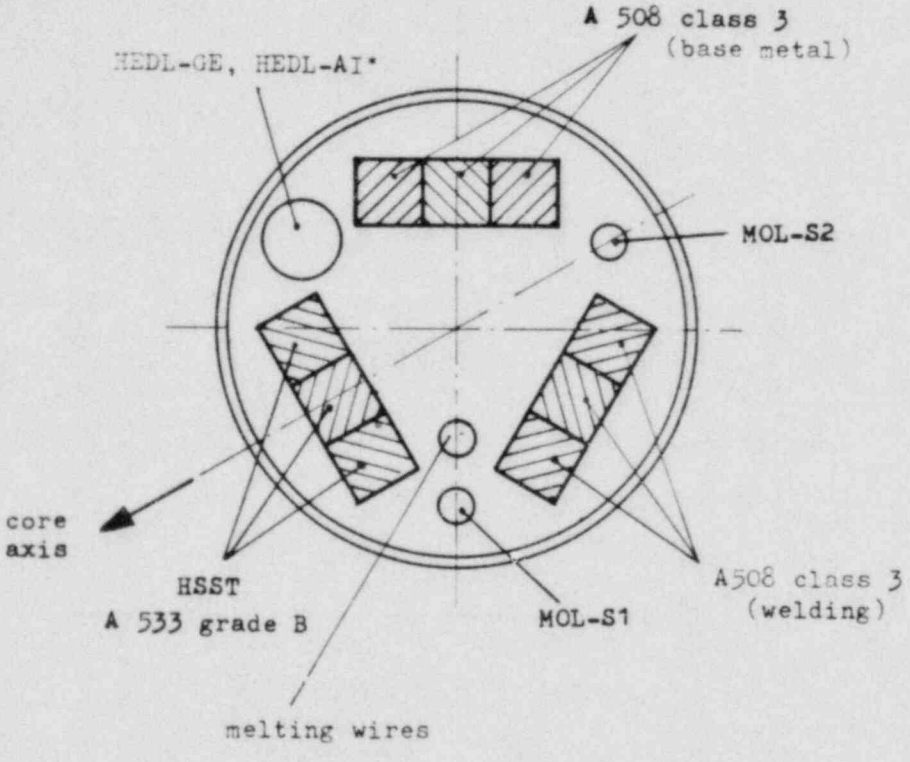


Fig. 4a : Positions of charpy's and dosimeters in LF positions 29 and 63 (+ 695 mm)

*The total available length (110 mm) is split equally between HEDL-GE and HEDL-AI dosimeter capsules

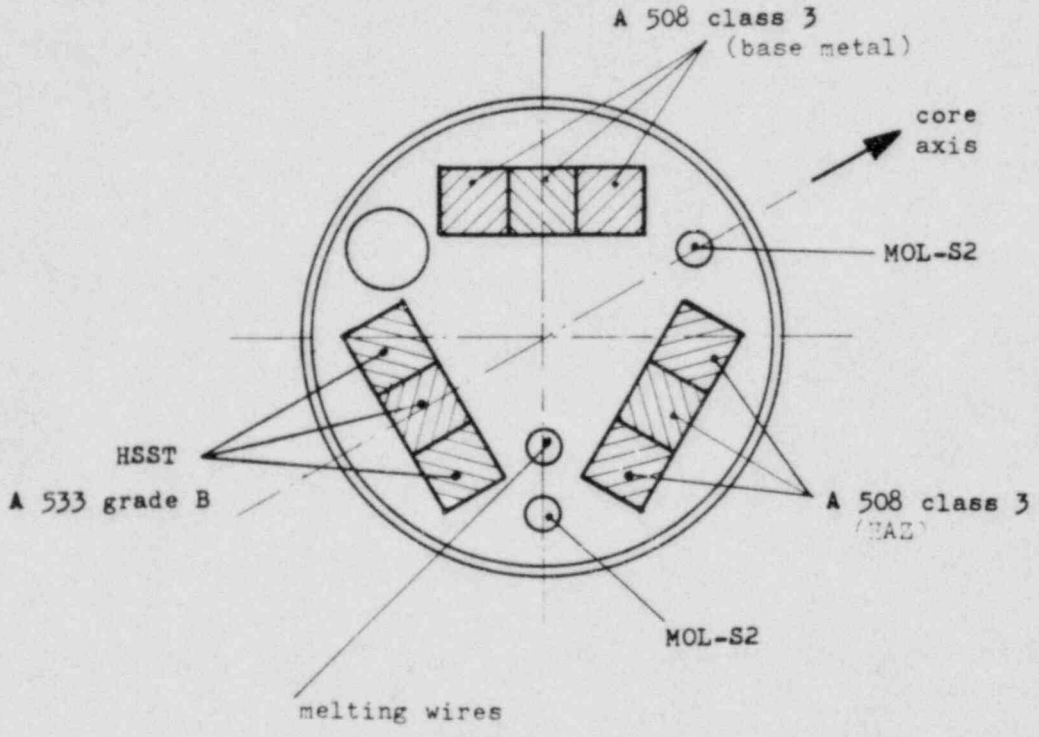


Fig. 4b : Positions of charpy's and dosimeters in LF positions 8 and 84 (+ 695 mm)

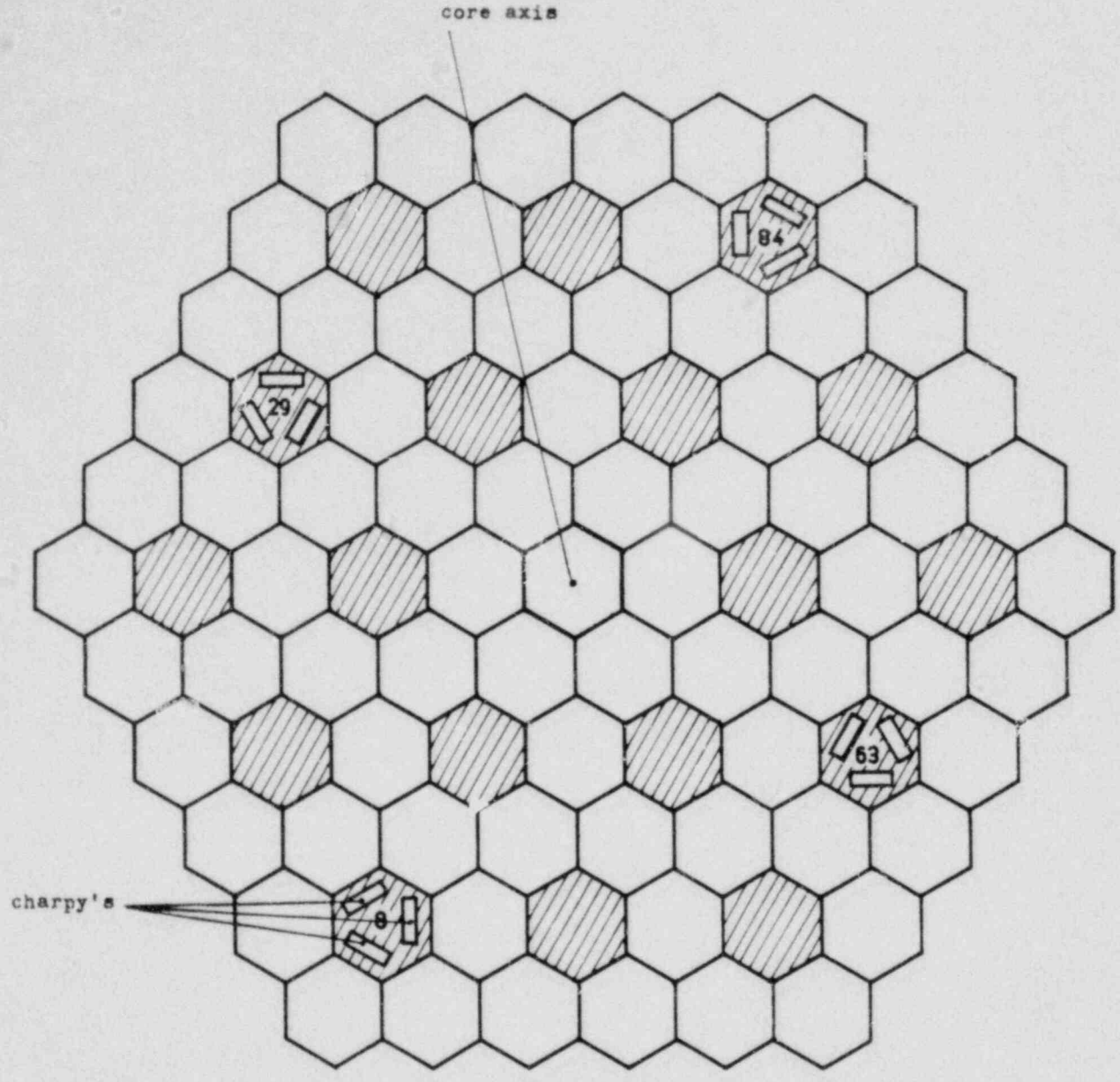


Fig. 5 : Overall view of the orientation of the charpy specimens with re ;ct to the BR3 core axis

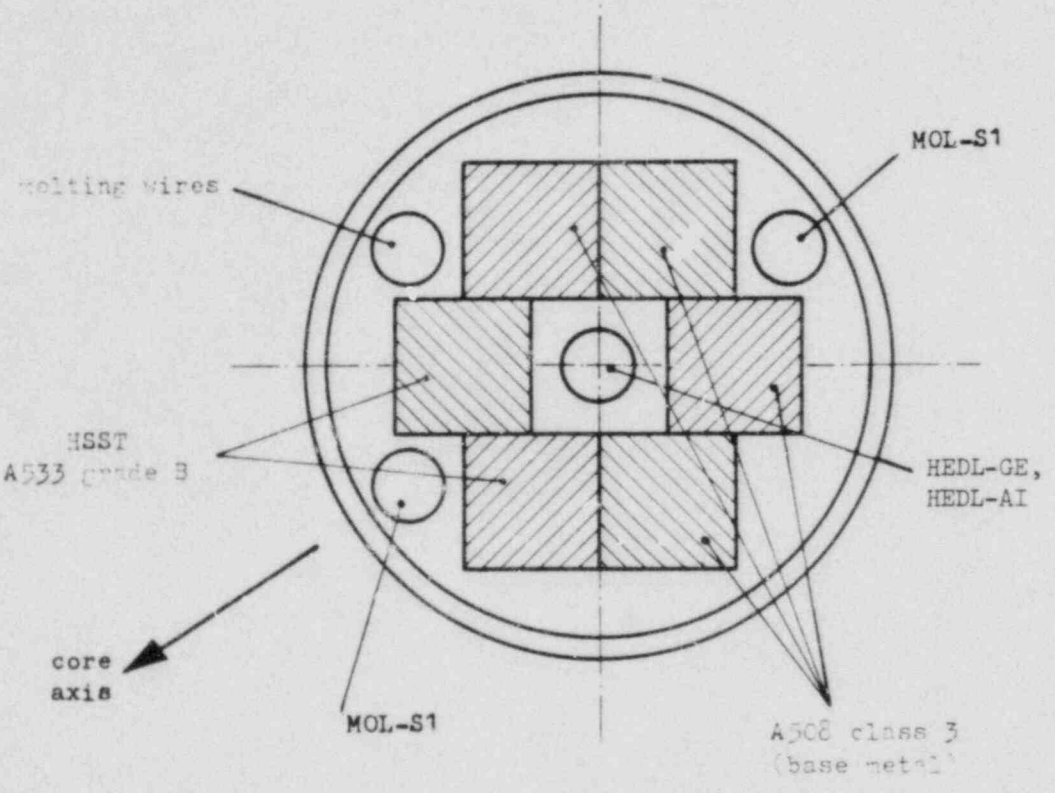


Fig. 6a : Positions of charpy's and dosimeters in the HF position 4 (+ 470 mm)

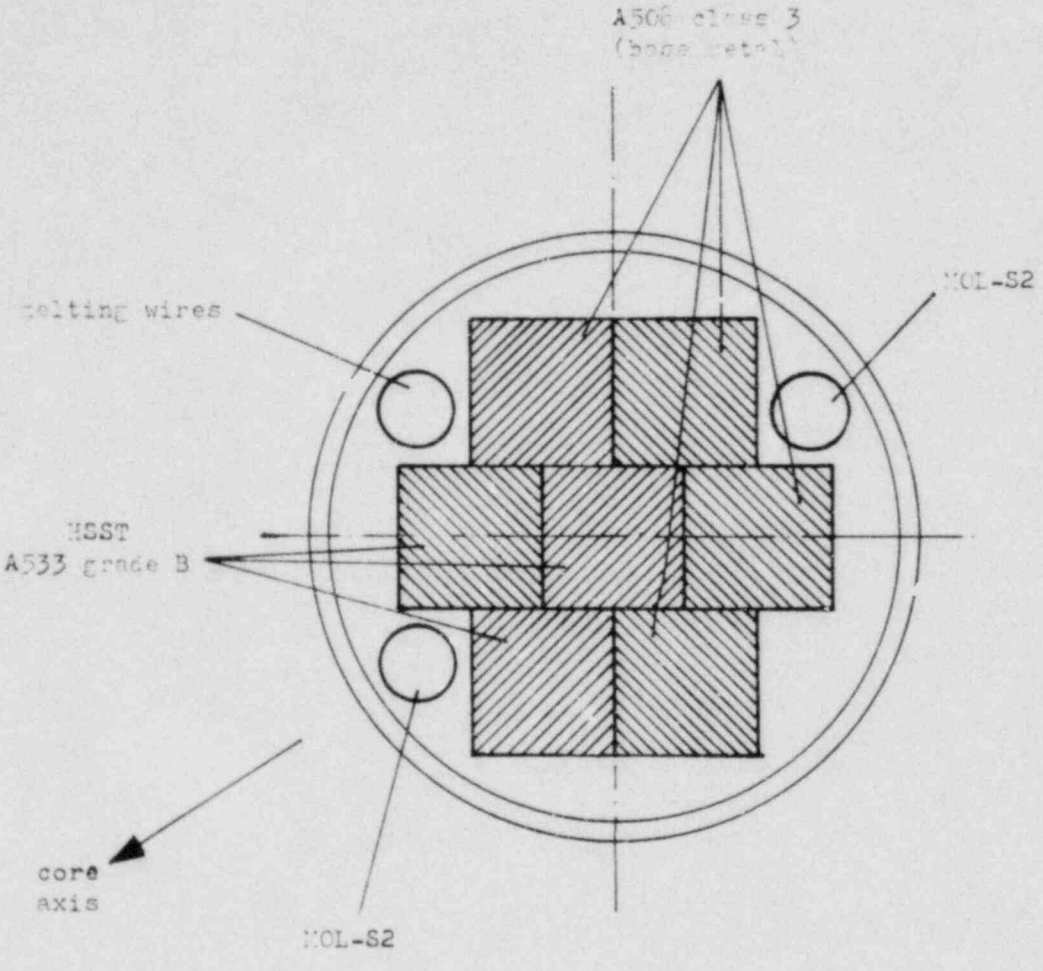
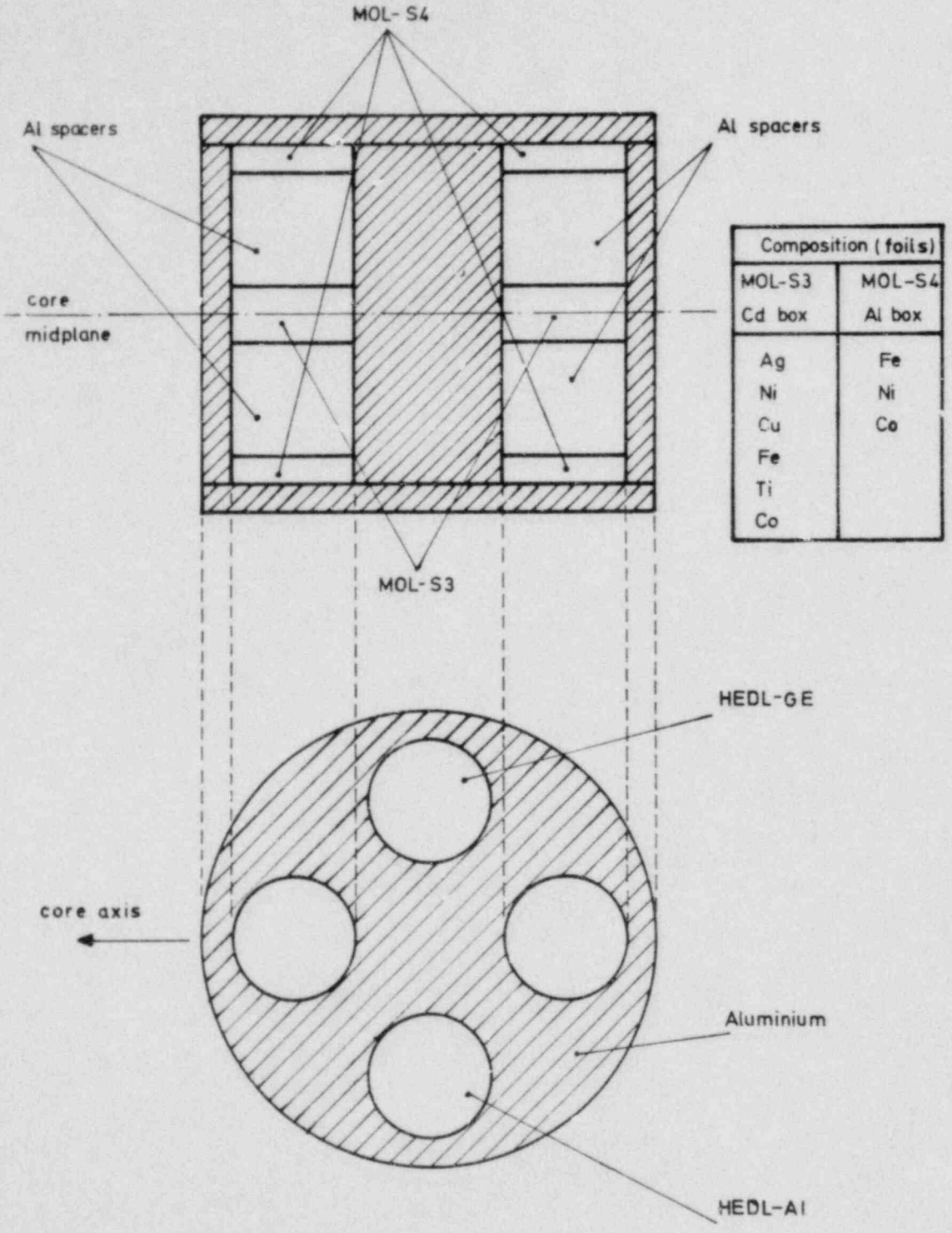
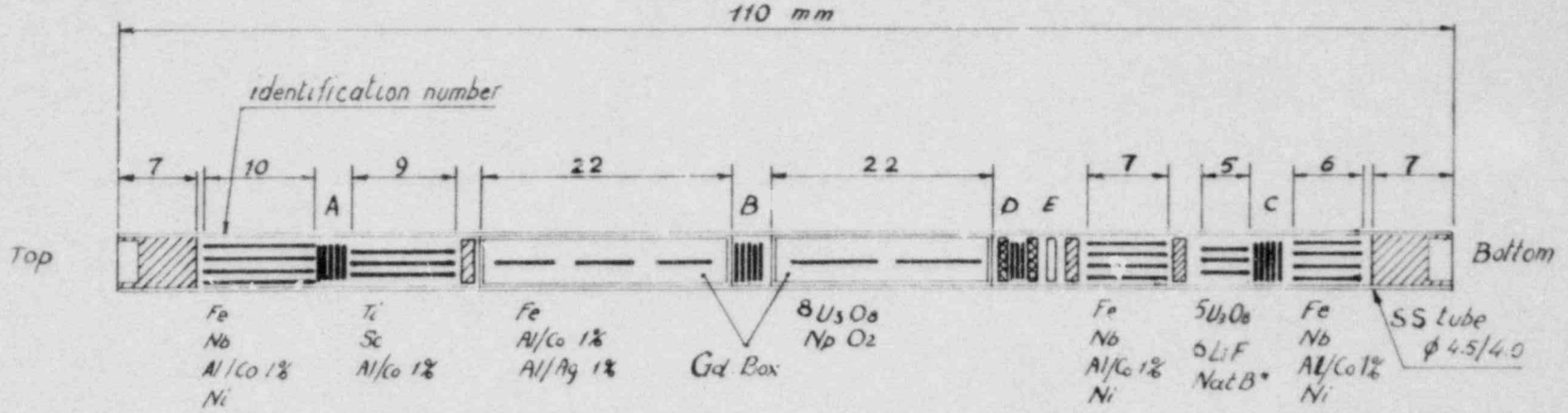


Fig. 5b : Positions of Charpy's and dosimeters in the HF position 4
(- 470 mm)



Composition (foils)	
MOL-S3 Cd box	MOL-S4 Al box
Ag	Fe
Ni	Ni
Cu	Co
Fe	
Ti	
Co	

fig 7 : Positions of the dosimeter capsules in position I (120°)



A.B.C.D : Cu, Fe and Nb* with spacers

E : quartz

▨ : Alluminium spacer

CEN/SCK-50

Fig. 8. MOL · S1 Set

* only in the cold leg 2 position (level 0 mm)

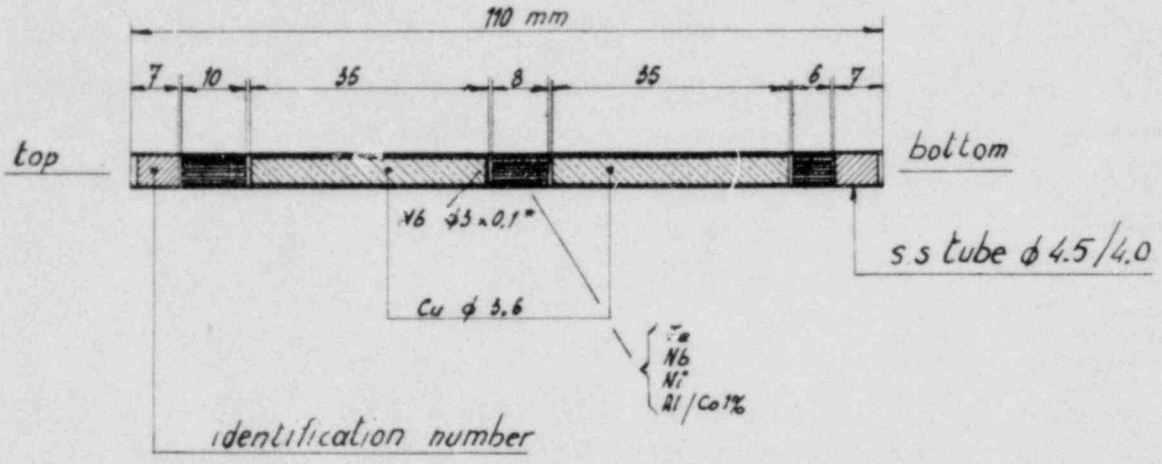


Fig 9 MOL-S2 set

* only in the cold leg 2 positions

F. Interface with Power Plant Surveillance : Dosimetry

J. DEBRUE, H. TOURWE, P. DE REGGE, A. FABRY

Objective

Provide information of interest from the dosimetry point of view as a result of the neutron fluence measurements performed by CEN/SCK in the frame of the pressure vessel surveillance programme for the DOEL and TIHANGE power plants.

Summary

The first capsule from DOEL I and DOEL II were unloaded respectively in February 1977 and in October 1977. After dismantling, the dosimeters were counted and analyzed. Reliable results are available at present for the non-fissile dosimeters from DOEL I; the reaction rate ratios are consistent with the calculated neutron spectrum provided by the plant operator, using the ENDF/B-IV cross sections for nickel and iron, and the Mann-Schenter data (HEDL) for copper. Technical difficulties were encountered in the manipulation and in the recovery of the fissile dosimeters (Np and U powders). Poor internal consistency has been achieved in the measurement of different fission products for neptunium; fluence values are significantly higher than those provided by the non-fissile dosimeters.

Accomplishment

CEN/SCK is responsible for the mechanical test on metallurgical specimens and for the dosimetry work at the request of the utilities operating the belgian power plants. Three pressurized water reactors, besides BR3, are at present in operation in Belgium

- DOEL I and DOEL II, two plants of 395 MWe each, started respectively in January 1975 and in November 1975

- TIHANGE I, 870 MWe, started in September 1975.

The pressure vessels of the DOEL reactors were manufactured by COCKERILL; the steel used is SOUDOTENAX 56, the composition of which is as follows (in wt %), for the base metal :

C 0.15 ; Co $\leq$ 0.02 ; Cr 0.6 ; Mn 1 ; Mo 0.35 ; Ni 0.8 ; P $\leq$ 0.015 ; Si 0.3 .

Irradiation tests on this type of steel were conducted in the high flux materials testing reactor BR2 (Mol) in 1965.

Six identical surveillance capsules have been loaded in DOEL I and in DOEL II. They contain each 9 tensiles, 5 WOL and 36 resilience specimens of SOUDONETAX 56, and 4 resilience specimens of a reference steel (A533B). They are located between the thermal shield and the pressure vessel at angular positions of 13°, 23° and 33° with respect to one of the core symmetry axes.

The arrangement of the neutron dosimeters in each capsule is shown on fig. 1. CEN/SCK was not involved in the design nor in the procurement of the dosimeters. There are in total four Cu wires, one Ni wire, thirteen 0.1 % Co/Al wires (three of which under cadmium), one uranium and one neptunium power sample. Each of these samples is contained in respectively a small brass (uranium) and a stainless steel (neptunium) box, wrapped in cadmium oxide.

Detailed specifications for the uranium and neptunium dosimeters could not be obtained up to now. Quality assurance documents are available for the cobalt, nickel and copper wires only and also for the melting wires. The quality assurance documents and the corresponding materials were delivered by CBNM-Geel.

Measurement of the non-fissile dosimeters from DOEL I (first surveillance capsule)

The first surveillance capsule was unloaded after two years of reactor operation, from January 1975*) to February 1977. The fast neutron fluence was first determined on the basis of the $^{58}Ni(n,p)^{58}Co$ and $^{63}Cu(n,\alpha)^{60}Co$ reaction product activities in the nickel and copper dosimeters.

---

*) The reactor was started in July 1974 but the regular operation at full power began in January 1975.

The activity of remnants from the metallurgical specimens, due to the reaction $^{54}Fe(n,p)^{54}Mn$, was also measured.
The theoretical spectrum between 0.625 eV and 10 MeV, provided by the plant operator, was extended up to 18 MeV, assuming a Watt spectrum in the 10 - 18 MeV interval, in order to calculate average cross sections over the range 111 keV - 18 MeV, thus roughly above 0.1 MeV.
The following results were obtained (all results in millibarn) :

$^{58}Ni(n,p)$	39.1	With ENDF/B-IV differential cross sections
$^{54}Fe(n,p)$	28.8	
$^{63}Cu(n,\alpha)$	0.161	
$^{63}Cu(n,\alpha)$	0.228	With cross-section calculated by W. Mann and R. Schenter (HEDL)

The determination of the neutron fluence above 111 keV, using these cross sections and the measured activities, showed a better consistency when accepting the value of 0.228 mb for copper :

$\frac{\text{Ni fluence}}{\text{Cu fluence}}$	$\frac{\text{Fe fluence}}{\text{Cu fluence}}$	
0.722	0.664	With ENDF/B-IV cross sections
1.025	0.942	With ENDF/B-IV cross sections for Ni and Fe, and Mann-Schenter data for Cu

The variation of the fast flux over the length of the capsule is less than 10 %. The nickel and copper wires were aligned verti ally : the results are believed to be independent of the flux gradient across the capsule; the steel remnants however came from unknown radial locations inside the capsule. Error estimates are therefore : $\pm$ 3 % for the Ni/Cu activity ratio and $\pm$ 5 % for the Fe/Cu activity ratio.
It was assumed that the time variation of the reactor power was representative for the fast flux variation at the capsule location.

The absolute fluence values are the following :

> 100 keV $1.7\ 10^{19}$ n/cm^2

> 1 MeV $6.1\ 10^{18}$ n/cm^2

They will be considered as preliminary as long as all the results from the first DOEL I and DOEL II capsules, including the fissile dosimeter results, will not be available.

The 0.1 % Co/V dosimeters gave a thermal neutron fluence $nv_o t = 1.1\ 10^{19}$ n/cm^2 in the center of the capsule.

Measurement of the fissile dosimeters from DOEL I

- The activity of the uranium containing brass box was very high by reason of the activation of silver present in the weld. The manipulation took place in a hot cell where the risk of contamination was not negligible. The fission product activities (^{106}Rh, ^{137}Cs, ^{144}Ce) in the ^{238}U dosimeter were found much higher than can be expected on the basis of the epithermal and fast neutron fluences (as determined with the non-fissile dosimeters) and of the measured isotopic composition of the uranium.
  More information will be reported when the DOEL II results will be available. An improved procedure is adopted to prevent the risk of contamination.

- The neptunium dosimetry was handled more easily. The neptunium powder and the brass box were dissolved together. The neptunium content in the solution was determined by alpha spectroscopy of ^{237}Np and by gamma spectrometry of ^{233}Pa (decay product of ^{237}Np); both results agreed within 4 % ; the value obtained is 11.0 mg. By reason of the low neutron exposure, the number of fissions could not be determined through the amount of ^{148}Nd produced. Three fission products were identified by gamma spectrometry : ^{106}Rh (decay product of ^{106}Ru), ^{137}Cs and ^{144}Ce . The results are summarized in the table

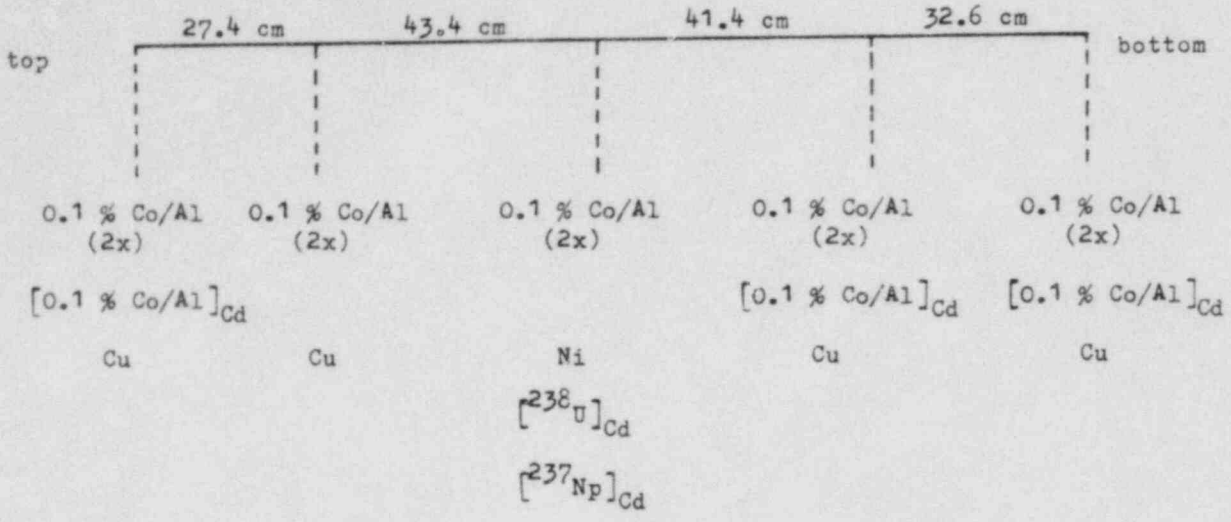
Isotope	Half-life	Activity[a] (μCi)	Correction[b] factor	F.P. yield (%)	Number of fissions	Neutron fluence[c] > 100 keV (n/cm^2)
^{106}Ru	368 d	4.56	1.879	1.56	9.3×10^{14}	$3.8\ 10^{19}$
^{137}Cs	30 y	1.14	1.024	5.9	1.0×10^{15}	$4.1\ 10^{19}$
^{144}Ce	284 d	9.73	2.188	4.4	6.3×10^{14}	$2.6\ 10^{19}$

a) at the end of the irradiation period : 19.2.1977

b) taking into account the irradiation history

c) calculated with ^{237}Np cross section from ENDF/B-IV, using the calculated neutron spectrum.

There appears an internal inconsistency in the fission product results and a significant discrepancy between these values and the non-fissile dosimetry data. An error analysis will be carried out when the DOEL II data will be available.



Melting wires at each position : 97.5 % Pb - 2.5 % Ag (304°C)
97.5 % Pb - 1.75% Sn - 0.75 % Ag (310°C)

Fig. 1 : Arrangement of the neutron dosimeters in the surveillance capsules

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NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

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1. REPORT NUMBER (Assigned by DDC)
NUREG/CR-1241, Vol. 4
HEDL-TME 80-6

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
LWR Pressure Vessel Irradiation Surveillance Dosimetry
Quarterly Progress Report
October 1980 - December 1980

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
G.L. Guthrie, W.N. McElroy

5. DATE REPORT COMPLETED
MONTH: August YEAR: 1981

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Hanford Engineering Development Laboratory
P.O. Box 1970
Richland, WA 99352

DATE REPORT ISSUED
MONTH: November YEAR: 1981

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.
FIN B5988

13. TYPE OF REPORT
Quarterly Progress Report

PERIOD COVERED (Inclusive dates)
October 1980 - December 1980

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

The Light Water Reactor Pressure Vessel (LWR-PV) Surveillance Dosimetry Improvement Program has been established by NRC in recognition of the importance of improving, maintaining, and standardizing neutron dosimetry, damage correlation, and the associated reactor analysis data and 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 USNRC supported activities at HEDL, ORNL, NBS, and NRL 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 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. Several subcommittees of ASTM E10 are responsible for the preparation of LWR-PV surveillance standards.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

21. NO. OF PAGES

20. SECURITY CLASS (This page)
Unclassified

22. PRICE
$

NRC FORM 335 (7-77)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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