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LWR PRESSURE-VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

QUARTERLY PROGRESS REPORT
APRIL 1982 - JUNE 1982

Hanford Engineering Development Laboratory

Prepared by
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Prepared for the U.S. Nuclear Regulatory Commission

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Hanford Engineering Development Laboratory

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PREVIEWS REPORTS IN LWR-PV-SDIP SERIES

NUREG/CR-0038	HEDL-TME 78-4	July - September 1977
NUREG/CR-0127	HEDL-TME 78-5	October - December 1977
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NUREG/CR-1241 Vol. 3	HEDL-TME 80-6	October 1980 - December 1980
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NUREG/CR-2345, Vol. 4	HEDL-TME 81-36	October 1981 - December 1981
NUREG/CR-2805, Vol. 1	HEDL-TME 82-18	January 1982 - March 1982

*Annual Reports

FOREWORD

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

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

These studies will establish and certify the precision and accuracy of the measurement and predictive methods recommended in the ASTM Standards and used for the assessment and control of the present and end-of-life (EOL) condition of pressure vessel and support structure steels. Consistent and accurate measurement and data analysis techniques and methods, therefore, will be developed, tested and verified along with guidelines for required neutron field calculations used to correlate changes in material properties with the characteristics of the neutron radiation field. It is expected that the application of the established ASTM Standards will permit the reporting of measured materials property changes and neutron exposures to an accuracy and precision within bounds of 10 to 30%, depending on the measured metallurgical variable and neutron environment.

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

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

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

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

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

CONTENTS

TABLES

ACRONYMS

ASTM	American Society for Testing and Materials
BSR	Bulk Shielding Reactor
BWR	Boiling Water Reactor
CEN/SCK	Centre d'Etudes Nucleaires de Saclay (France)
ENSA	Engineering Services Associates
EPRI	Electric Power Research Institute
FBR	Fast Breeder Reactor
FSAR	Final Safety Analysis Review
HEDL	Hanford Engineering Development Laboratory
KFA	Kernforschungsanlage (Jülich, Germany)
LWR	Light Water Reactor
MEA	Materials Engineering Associates
MFR	Magnetic Fusion Reactor
NBS	National Bureau of Standards
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Research Reactor (ORNL)
PCA	Poolside Critical Assembly
PSF	Poolside Facility (ORNL)
PV	Pressure Vessel
PWR	Pressurized Water Reactor
QA	Quality Assurance
RI	Rockwell International
SDIP	Surveillance Dosimetry Improvement Program
SDMF	Simulated Dosimetry Measurement Facility
SEM	Scanning Electron Microscope
SRM	Standard Reference Material
SSC	Simulated Surveillance Capsule
SSTR	Solid State Track Recorder
TLD	Thermoluminescent Dosimeter
UK	United Kingdom
WRSR	Water Reactor Safety Research

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The following organizations are presently participating 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 (AERE-H), Harwell, UK

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

Engineering Services Associates (ENSA), USA

Fracture Control Corporation (FCC), USA

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

Hanford Engineering Development Laboratory (HEDL), USA

Institüt für Kernenergetik und Energiesysteme der Universitat Stuttgart (IKE), Stuttgart, Germany

IRT Corporation (IRT), USA

Italian Atomic Power Authority (ENEL), Italy

Japan Atomic Energy Research Institute (JAERI), Japan

Kernforschungsanlage Jülich Gmbh (KFA), Germany

Kraftwerk Union, Germany

Materials Engineering Associates (MEA), USA

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

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 (W-NTD), USA

Westinghouse Electric Corporation - Research and Development Division (W-R&D), USA

SUMMARY

HANFORD ENGINEERING DEVELOPMENT LABORATORY (HEDL)

A brief program status report is presented with a list of planned NUREG reports that addresses individual and combined PWR and BWR physics-dosimetry-metallurgy issues. They will provide a reference base of information to support the preparation of new set of LWR ASTM Standards (Figures S-1 and S-2).

A least squares computer code has been developed that minimizes the total weighted sum of squares of the residuals in both the Charpy shift and the logarithm of the fluence, when applied to the problem of fitting a trend curve to Charpy shift data for irradiated surveillance specimens. The new code calculates an unbiased estimate for the exponent of the fluence in simple laws where the Charpy shift is assumed to be proportional to the fluence raised to a power. This feature is an improvement over previously used codes for least squares fits of this general type. The most recent improvement in the code is that it has been modified to require complete correlation in the fluence adjustments for specimens irradiated in a single capsule. Several functional forms have been used with the revised code, including one discussed in the previous quarterly report (NUREG/CR-2805, Vol. 1, HEDL-TME 82-18). One of the forms investigated in the present report uses a fluence exponent that is a slowly varying function of the fluence. Some of the other forms investigated have the features that limit the incremental contribution of Ni at high Ni levels, for low value of the Cu concentration. Some improvement has been found in the standard deviation for the fit, compared to the standard deviation previously reported.

OAK RIDGE NATIONAL LABORATORY (ORNL)

The coupled neutron-gamma calculations for the PCA 12/13 configuration has been completed. The conclusions indicate that in general the revised coupled calculations of Minsart are confirmed, and that a careful analysis of the ^{235}U (n,f) reaction rates show agreement with all reported measurements.

The final cumulative irradiation and temperature distribution data (Tables ORNL-5, 6, and 7) and the reactor power time history data (Table ORNL-8) are reported for all the LWR-PVS capsules in the ORR-PSF.

The B&W surveillance capsules perturbation experiment is ready to be irradiated.

The counting of all the dosimeters from Capsule C of the fourth HSST irradiation series has been completed.

The status of the three ASTM standards for which ORNL has the lead is as follows:

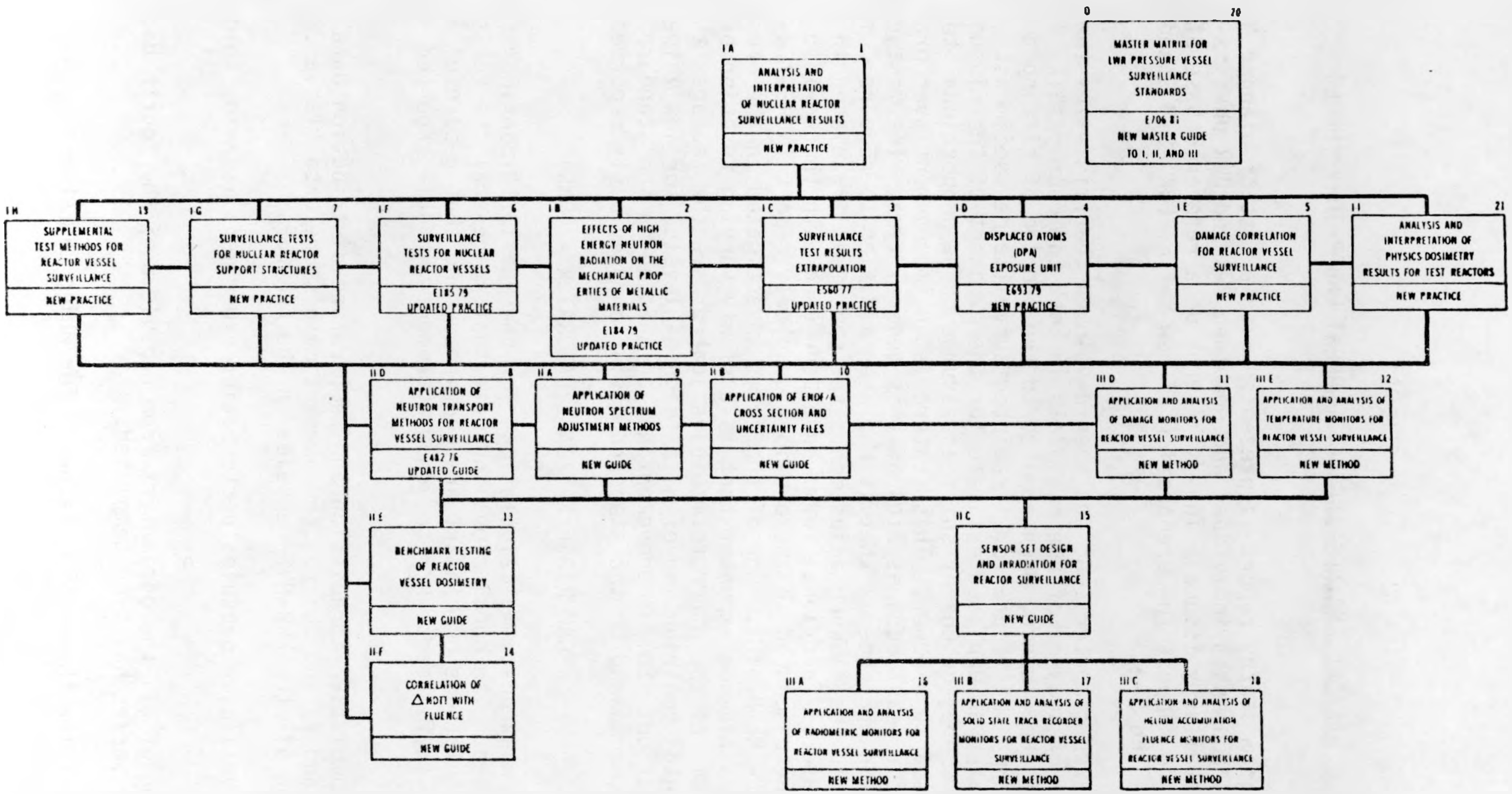
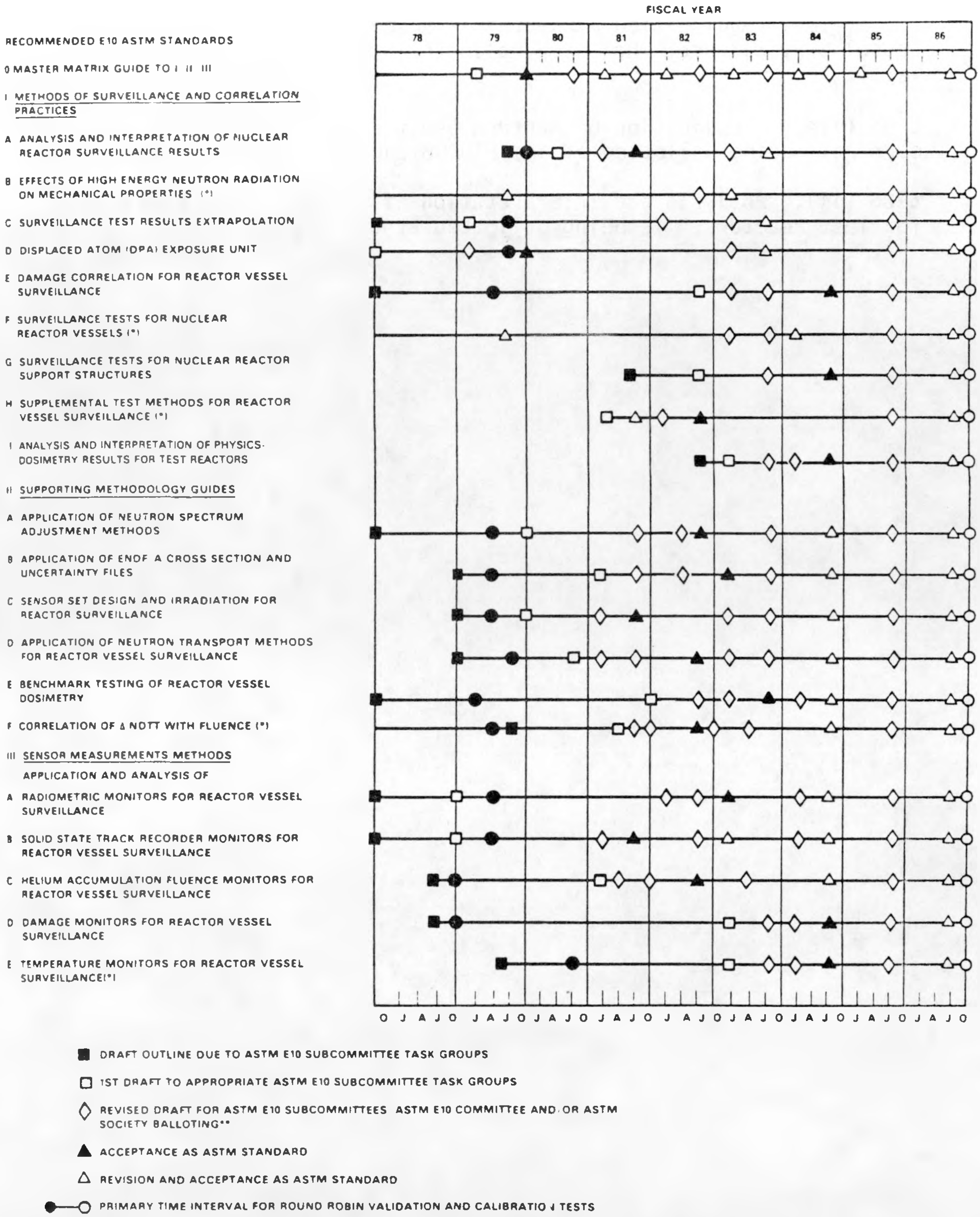


FIGURE S-1. ASTM Standards for Surveillance of LWR Nuclear Reactor Pressure Vessels and Support Structures.



*AN ASTERISK INDICATES THAT THE LEAD RESPONSIBILITY IS WITH SUBCOMMITTEE E10.02 INSTEAD OF WITH SUBCOMMITTEE E10.05
**THE 1985-1986 REVISIONS WILL PRIMARILY ESTABLISH STANDARD-TO-STANDARD SELF CONSISTENCY

FIGURE S-2. ASTM LWR Standards Preparation Schedule.

a. E706 (IID), "Application of Neutron Transport Methods for Reactor Vessel Surveillance," has been ballotted and approved at the Society level.

b. E706 (IIA), "Application of Neutron Spectrum Adjustment Methods," is currently being ballotted at the E10.05 and E10 levels simultaneously.

c. E706 (II), "Analysis and Interpretation of Physics--Dosimetry Results for Test Reactors," is being ballotted at the E10.05 level.

Hanford Engineering Development Laboratory

(HEDL)

DEVELOPMENT OF TREND CURVE FORMULAS USING SURVEILLANCE DATA-II

G. L. Guthrie - HEDL

Objective

The objective of the present work is to develop formulas relating the irradiation induced shift in nil ductility transition temperature (30 ft-lb Charpy), the irradiation fluence, and the nickel and copper concentrations of pressure vessel steel surveillance specimens. This work is an extension of work reported in NUREG/CR-2805, Vol. 1, HEDL-TME 82-18.[1]

The establishment of trend curve formulas is pertinent to the writing of ASTM standards on 1) ΔNDTT vs fluence and 2) damage correlation. These standards are required as part of the LWR PV Surveillance Dosimetry Improvement Program.

Summary

A least squares computer code has been developed that minimizes the total weighted sum of squares of the residuals in both the Charpy shift (measured minus calculated) and the logarithm of the fluence

$$\sum \left[\log_e \left(\frac{\text{measured fluence}}{\text{best adjusted fluence}}\right)\right]^2$$

The new computer code produces an unbiased estimate of the fluence exponent in simple laws of the type

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

where:

ΔT = Irradiation-induced Charpy shift (30 ft-lb)
A = Chemical factor
ϕt = Irradiation fluence (n/cm^2, E > 1.0 MeV)

The most recent improvement in the code is that the fluence adjustments are now restrained to require complete correlation for all exposures in a single surveillance capsule. The code has been used to process data supplied by Dr. P. N. Randall of NRC.

Several functional forms have been used with the revised code, including the form discussed in the previous report. One of the forms used in the present report uses an exponent which is a slowly varying function of fluence. Some other functional forms investigated in the present report have been chosen so as to have the feature that they limit the contribution attributed to Cu·Ni interactions at high values of the Cu·Ni product, or at high nickel levels. Small improvements are found in the standard deviation, compared to that reported in the previous quarterly.[1] In general, the influence of the CuNi interaction term is less than previously found.

Accomplishments and Status

Additional modifications have been made in the nonlinear least squares code used for the trend curve work discussed in the previous report.[1] The basic feature that distinguishes this code from most other nonlinear codes is that it has the ability to consider errors in the reported fluence when making parameter adjustments in Charpy trend-curve laws.

The code minimizes the quantity

$$SSE = \sum_i \left(\Delta T_{im} - \Delta T_{i\ calc}\right)^2 + W \sum_i \left[\log_e (\phi t)_{i\ M} - \log_e (\phi t)_{i\ true}\right]^2 \tag{2}$$

Where:

ΔT_{im} = Measured value of the shift in the 30 ft-lb Charpy transition temperature for the ith data point

$\Delta T_{i\ calc}$ = Calculated value of the same quantity

$(\phi t)_{i\ M}$ = Measured fluence (n/cm^2, $E > 1.0$ MeV)

$(\phi t)_{i\ true}$ = Adjusted value of the fluence, adjusted to give the best value in a least squares sense in Eq. (2). (The values of $\Delta T_{i\ cal}$ are calculated using $(\phi t)_{i\ true}$ in the functional form being adjusted.)

W = Relative weight of the errors in the logarithms of the fluence, compared to the Charpy measurement errors. This relationship was discussed in the previous report.(1)

The complete set of adjustable parameters consists of the usual parameters in any chosen ΔT relationship, plus all the fluence values.

The newly developed code (used in Ref. 1) has the feature that it produces unbiased estimates for N, the fluence exponent, in relations of the type $\Delta T \propto (\phi t)^N$.

The most recent modification of the code restricts the fluence adjustments to require complete correlation between adjustments of the fluence for exposure values of specimens in a single capsule.

The revised code has been applied to the data supplied by Dr. P. N. Randall of NRC. Several functional forms have been used, and the code has found the best values (in a least squares sense) for the adjustable parameters and the irradition fluences. Several of the individual investigations are described below.

As described in the previous report,[1] the data consisted of 139 data points containing the copper concentration, fluence, 30 ft-lb Charpy shift, and the nickel concentration or an estimate thereof. Two sets of fluence values n/cm^2 (E > 1.0 MeV) were available: 1) a set supplied by Randall, and a revised set based on the latest work of R. L. Simons.*[2,3] Three of the data points, namely numbers 10, 13, and 68 were suspect on a 3 σ deviation basis, as judged from earlier work using a conventional nonlinear least squares technique. Individual least squares fits were performed with 136 data points, 139 data points, and amounts in between, as one or more of these three points were omitted in the separate runs.

The complete data set is shown in Table HEDL-1.

The general form being fitted was

$$\Delta T = f_1 \text{ (chemistry)} \cdot \text{(exposure)}^{f_2} \cdot \left\{ \left[(1 + TC \cdot (550°\text{F-Irr Temp}) \right] \right\} \qquad (3)$$

where TC is the adjustable temperature coefficient, Irr Temp the irradiation temperature, and f_1 the pre-multiplier used as a function of chemistry that contained in all the cases an adjustable additive constant, a Cu term (linear) and a Cu$\cdot$Ni interaction term.

The Cu$\cdot$Ni interaction term took various forms of 1) Cu$\cdot$Ni; 2) Cu$\cdot$tanh (x$\cdot$Ni/Cu), where x is an adjustable parameter; 3) Cu$^2\cdot$tanh(x$\cdot$Ni/Cu); 4) square root of (Cu$\cdot$Ni); 5) x$\cdot$Cu$\cdot$Ni + y$\cdot$Cu$\cdot$Ni^2, where x and y are adjustable parameters; and 6) Cu^2/Ni. In the temperature correction term, for most of the runs, the TC factor was set to zero. The exposure term used units of dpa or fluence (n/cm^2, E > 1.0 MeV) as supplied by R. L. Simons or P. N. Randall. The exposure exponent, f_2, was assumed to be either an adjustable constant or a linear function of $\log_e(\phi t)$. This latter form allowed the exponent to be a slowly varying function of the exposure. This form dispenses with the assumption that log(ΔT) vs log(fluence) plots as a straight line.

The rationale behind some of the "f_1" Cu$\cdot$Ni interaction terms is the following. (A), as was related in Ref. 1, J. R. Hawthorne[4] and others have reported that Ni is relatively innocuous for irradiation embrittlement in the presence of very low copper levels, but causes irradiation embrittlement at high levels of Cu. (B) G. R. Odette and others suggest that there may be a limit to the incremental embrittlement caused by additional Ni after the Ni/Cu ratio gets beyond some fixed level. Form (1) is responsive to comment (A).

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*These sets are referred to as (PNR>1) and (RLS>1) in the listing in Table HEDL-2.

At low levels of Ni/Cu, Form (2) can be expanded in a Taylor's series to give a value proportional to Ni at low levels of Ni/Cu; but for any fixed Cu level, the total contribution is limited at high levels of Ni/Cu. Thus, this formulation is in accord with comments (A) and (B). Formulation (3) gives a contribution proportional to Cu•Ni at low levels of Ni/Cu, but saturates at higher levels of the Ni/Cu ratio. For form (4), the properties of the square root of (Cu•Ni) are obvious and are in accord with suggestions (A) and (B). Form (5) has the desired qualities if x is positive and y is negative, for a range of values of Cu and Ni, and form (6) is better understood by regarding it as having arisen from a form of the type $Ni \cdot (x \cdot Cu/Ni) + y \cdot (Cu/Ni)^2$). This reduces to $x \cdot Cu + y \cdot Cu^2/Ni$, and the Cu term would be absorbed into the standard linear Cu term present elsewhere in the formula. This formulation allows a Ni contribution that is a nonlinear function of the (Cu/Ni) ratio.

As was discussed in Ref. 2, there is great need for a mechanistic understanding of the processes involved, and the experimental techniques available do not provide sufficient opportunities for acquiring such information. In the absence of a well established model, any relations derived by statistical methods should be applied with great caution in regions of independent variables outside the range of the data used to determine the parameters.

The results of 16 separate least squares fits are shown in Table HEDL-2.

In computer run number 1, the omission of points 121 and 122 was due to a lack of knowledge of the dpa exposure value for the particular reactor. In the columns giving details of the makeup of "f_1", the entry "tanh" refers to a factor "tanh(X•Ni/Cu)," where x is an adjustable constant.

The relative weight "W" was calculated by taking the ratio

$$W = \left\{ \frac{\delta\ [\Delta T(°F)]}{\log_e\ (1 + \text{fractional uncertainty in fluence})} \right\}^2 \qquad (4)$$

where: δ (ΔT(°F)) is the uncertainty in a measured Charpy shift values. For a 15°F uncertainty in ΔT and a 35% uncertainty in fluence, W = 2498. For a 20°F uncertainty in ΔT and a 25% uncertainty in fluence, W = 8033.

Several conclusions can be drawn from Table HEDL-2. Comparing runs 6 and 7, there was a noticeable but not overwhelming improvement in using the latest exposure values supplied by R. L. Simons. Comparing Runs 1 and 2, the use of dpa in place of fluence made essentially no difference at all. This might be expected since nearly all the data came from surveillance capsules where the spectral shapes were quite similar. Increasing the weight factor W reduced the calculated value of the standard deviation, as can be seen by comparing runs 9 and 11. This is somewhat of an artifact. The standard deviation in the table was calculated as follows:

The least squares program adjusted both the parameters (in the Charpy relation) and the fluence values, to minimize SSE of Eq. (2), for the relationship being investigated. The standard deviation was calculated using the "best" parameter values. The sum of the squares of residuals used in the standard deviation contained only the residuals between the measured and calculated Charpy shift, with the calculated values determined using the measured values of the fluence rather than the best adjusted fluence values. Use of an artificially high W would produce parameter values more compatible with the measured exposures, and thus produce a low sum of squares of errors in the calculation of the standard deviation. However, this would ignore the bias produced by the incorrect parameter values, including that produced by the false low value of the exposure exponent. A proper estimate of sigma and the parameters requires the use of the best available estimate for W.

In Table HEDL-2, for the relations utilizing a temperature factor in the "shift" formula, the irradiation temperature was assumed to be 550°F for Westinghouse plants, 586°F for Babcock and Wilcox plants, and 568°F for Combustion Engineering plants.

For fit number 2 in Table HEDL-2, the complete formula and adjustable parameter values are given below.

$$\Delta T = [x(1) + x(2) \cdot Cu + x(3) \cdot Cu \cdot \mathrm{Tanh}(x(4) \cdot Ni/Cu)] \cdot \left(\frac{\phi t}{10^{19}}\right)^{\left[x(5) + x(6) \cdot \log_e\left(\frac{\phi t}{10^{19}}\right)\right]} \quad (5)$$

x(1) =	-28.5	x(4) =	0.277
x(2) =	521.	x(5) =	0.262
x(3) =	449.	x(6) =	-0.030847
		sigma =	20.31°F

Cu is to be entered in weight percent, and similarly for Ni. Fluence is to be entered as n/cm^2, (E > 1.0 MeV).

Acknowledgments

Much gratitude is owed to Neil Randall for supplying the data list and for consultation during the course of the work, Jerry Varsik for consultation and discussions relating to the importance of various alloying elements, G. R. Odette for discussions and suggestions regarding embrittlement trends of Cu·Ni and their interaction mechanisms and relative merits of various functional forms, R. L. Simons for providing improved fluence and dpa estimates, and R. Dierckx and F. W. Stallmann for discussions relating to the importance of adjusting the fluence values.

Expected Accomplishments

It is expected that a covariance matrix will be derived for the parameters of one or more of the formulas developed in the work being reported here. This can be used together with composition and exposure uncertainties to derive improved estimates of the probable error in the calculated Charpy shift.

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TABLE HEDL-1

CHARPY DATA USED IN ANALYSIS

Plant No.	Plant and Capsule	Material	Ni (%)	Cu (%)	Fluence (n/cm^2, E > 1.0 MeV) Randall	Simons	dpa (Simons)	Measured Shift (°F) in Charpy 30 ft-lb Temperature
1	MAINE	A533B P.L.	0.59	0.15	6.72 E18	6.66 E18	0.01	97.
2	YANKEE	A533B P.T.	0.59	0.15	6.72 E18	6.66 E18	0.01	93.
3	263	WELD	0.780	0.360	6.72 E18	6.66 E18	0.01	222.
4	MAINE	A533B P.L.	0.59	0.15	1.3 E19	2.1 E19	0.0356	120.
5	YANKEE	A533B P.T.	0.59	0.15	1.3 E19	2.1 E19	0.0356	120.
6	1st	WELD	0.78	0.36	1.3 E19	2.1 E19	0.0356	270.
7	ACCEL	HSST-01	0.71	0.18	1.3 E19	2.1 E19	0.0356	150.
8	FORT	A533 P.	0.48	0.10	5.1 E19	6.13 E18	0.00942	60.
9	CALHOUN	WELD	0.60	0.35	5.1 E19	6.13 E18	0.00942	238.
10	W225	HSST-01	0.71	0.18	5.1 E19	6.13 E18	0.00942	124.
11	PALISADES	A302B P.L.	0.53	0.25	4.5 E19	5.96 E19	0.0975	205.
12	1ST ACCEL	A302B P.T.	0.53	0.25	4.4 E19	5.96 E19	0.0975	205.
13	CAPSULE	WELD	0.53	0.26	4.4 E19	5.96 E19	0.0975	350.
14	MAINE	A533B P.L.	0.59	0.15	1.01 E20	8.73 E19	0.141	185.
15	YANKEE	A533B P.T.	0.59	0.15	1.01 E20	8.73 E19	0.141	195.
16	2ND ACCEL	WELD	0.78	0.36	1.01 E20	8.73 E19	0.141	345.
17	CALVERT	A533 P.	0.64	0.12	6.00 E18	6.0 E18	0.0093	60.
18	CLIFFS	WELD	0.18	0.24	6.10 E18	(default)	0.0093	59.
19	#1 263	HSST-02	0.68	0.14	5.90 E18	6.0 E18	0.0093	88.
20	3 MILE	A302B P.	0.57	0.09	1.07 E18	1.07 E18	0.00158	29.
21	ISLAND	WELD	0.71	0.34	1.07 E18	(default)	0.00158	117.
22	#1 Cap E	HSST-02	0.64	0.17	1.07 E18	1.07 E18	0.00158	44.
23	OCONEE	A533B P.L.	0.50	0.17	1.5 E18	1.65 E18	0.0022	53
24	#1	A533B P.T.	0.50	0.17	1.5 E18	1.65 E18	0.0022	32
25	Cap E	WELD	0.59	0.32	1.5 E18	1.65 E18	0.0022	124
26		HSST-02	0.64	0.17	1.5 E18	1.65 E18	0.0022	64
27	SAN	A302B P.	0.20	0.18	6.45 E19	5.09 E19	0.0944	120.
28	ONOFRE	WELD	0.20	0.19	6.45 E19	5.09 E19	0.0944	145.
29	#1 Cap F	ASTM 302 Cor	0.18	0.20	6.45 E19	5.09 E19	0.0944	130.
30	SAN	A302B P.-8	0.2	0.18	4.4 E19	4.08 E19	0.0705	110.
31	ONOFRE	-1	0.2	0.17	4.4 E19	4.08 E19	0.0705	140.
32	#1	-9	0.2	0.18	4.4 E19	4.08 E19	0.0705	130.
33	Cap D	ASTM 302 Cor	0.18	0.20	4.4 E19	4.08 E19	0.0705	150.
34	SAN	A302B P.-9	0.2	0.18	4.4 E19	2.53 E19	0.0424	100.
35	ONOFRE	WELD	0.2	0.19	4.4 E19	2.53 E19	0.0424	80.
36	#1 Cap D	ASTM 302 Cor	0.18	0.20	4.4 E19	2.53 E19	0.0424	115.

TABLE HEDL-1 (Cont'd)

Plant No.	Plant and Capsule	Material	Ni (%)	Cu (%)	Fluence (n/cm^2, E > 1.0 MeV) Randall	Simons	dpa (Simons)	Measured Shift (°F) in Charpy 30 ft-lb Temperature
37	HBR	A302B P.	0.20	0.10	7.76 E18	7.77 E18	0.0123	50.
38	#2	WELD	0.65	0.34	7.76 E18	7.77 E18	0.0123	200.
39	Cap V	ASTM 302 Cor	0.18	0.20	7.76 E18	7.77 E18	0.0123	95.
40	HBR	A302 -4	0.20	0.10	5.09 E18	5.07 E18	0.00858	30.
41	#1	-5	0.20	0.10	5.09 E18	5.07 E18	0.00858	30.
42	Cap S	-6	0.20	0.10	5.09 E18	5.07 E18	0.00858	20.
43		Cor Mon.	0.18	0.20	5.09 E18	5.07 E18	0.00858	80.
44	TURKEY PT	A508 CL2	0.68	0.079	1.99 E19	1.99 E19	0.0244	45.
45	#3	A508 CL2	0.70	0.058	1.99 E19	1.99 E19	0.0244	23.
46	Cap S	ASTM 302 Cor	0.18	0.20	1.99 E19	1.99 E19	0.0244	139.
47	TURKEY PT	A508 CL2	0.70	0.056	1.81 E19	1.75 E19	0.0241	11.
48	#4	A508 CL2	0.71	0.054	1.81 E19	1.75 E19	0.0241	35.
49	Cap S	HSST-02	0.68	0.140	1.81 E19	1.75 E19	0.0241	115.
50	PT BEACH	A508 CL2	0.70	0.051	2.56 E19	2.42 E19	0.0431	35.
51	#2	A508 CL2	0.71	0.088	2.56 E19	2.42 E19	0.0431	70.
52	Cap R	WELD	0.59	0.25	2.56 E19	2.42 E19	0.0431	230.
53		HSST-02	0.68	0.14	2.56 E19	2.42 E19	0.0431	151.
54	PT BEACH	A508 CL2	0.70	0.051	7.24 E18	7.48 E18	0.0122	20.
55	#2	A508 CL2	0.71	0.088	7.24 E18	7.48 E18	0.0122	30.
56	Cap V	WELD	0.59	0.25	7.24 E18	7.48 E18	0.0122	165.
57		HSST-02	0.68	0.14	7.24 E18	7.48 E18	0.0122	90.
58	PT BEACH	A508 CL2	0.70	0.051	1.04 E19	9.43 E18	0.0160	17.
59	#2	A508 CL2	0.71	0.088	1.04 E19	9.43 E18	0.0160	30.
60	Cap T	WELD	0.59	0.250	1.04 E19	9.43 E18	0.0160	145.
61		HSST-02	0.68	0.140	1.04 E19	9.43 E18	0.0160	105.
62	SURRY	A533B P.T.	0.54	0.110	3.02 E18	3.02 E18	0.00488	55.
63	#2	A533B P.T.	0.54	0.110	3.02 E18	(default)	0.00488	45.
64	Cap X	WELD	0.56	0.190	3.02 E18	3.02 E18	0.00488	95.
65		HSST-02	0.68	0.140	3.02 E18	(default)	0.00488	60.
66	KEWAUNEE	A508 123X	0.71	0.060	2.07 E19	2.07 E19	0.0366	15.
67	Cap R	A508 123X	0.75	0.060	2.07 E19	(default)	0.0366	20.
68		WELD	0.77	0.200	2.07 E19	2.07 E19	0.0366	235.
69		HSST-02	0.68	0.140	2.07 E19	(default)	0.0366	140.
70	KEWAUNEE	A508 122X	0.71	0.060	7.13 E18	6.6 E18	0.0116	0.
71	Cap V	A508 123X	0.75	0.060	7.13 E18	6.6 E18	0.0116	0.
72		WELD	0.77	0.200	7.13 E18	6.6 E18	0.0116	175.
73		HSST-02	0.68	0.140	7.13 E18	6.6 E18	0.0116	95.

TABLE 1 (Cont'd)

Plant No.	Plant and Capsule	Material	Ni (%)	Cu (%)	Fluence (n/cm^2, E > 1.0 MeV) Randall	Simons	dpa (Simons)	Measured Shift (°F) in Charpy 30 ft-lb Temperature
74	ARKANSAS	A533B P.L.	0.52	0.150	7.27 E17	7.27 E17	0.00103	19.
75	#1	A533BP.T.	0.52	0.150	7.27 E17	7.27 E17	0.00103	22.
76	Cap E	WELD	0.59	0.310	7.27 E17	7.27 E17	0.00103	137.
77		Cor. Mon.	0.64	0.170	7.27 E17	7.27 E17	0.00103	40.
78	COOK	A533B P.L.	0.49	0.140	1.8 E18	3.40 E18	0.00599	75.
79	#1	A533B P.T.	0.49	0.140	1.8 E18	3.40 E18	0.00599	75.
80	Cap T	WELD	0.74	0.270	1.8 E18	3.40 E18	0.00599	130.
81		Cor. Mon.	0.68	0.140	1.8 E18	3.40 E18	0.00599	70.
82	PRAIRIE	A508-CL3	0.72	0.060	7.0 E18	6.16 E18	0.0105	24.
83	ISLAND	WELD	0.17	0.130	7.0 E18	6.16 E18	0.0105	25.
84	#1 Cap V	HSST-02	0.68	0.140	7.0 E18	6.16 E18	0.0105	110.
85	PRAIRIE	A508 TAN	0.70	0.085	7.45 E18	6.86 E18	0.0118	35.
86	ISLAND	A508 TAN	0.70	0.085	7.45 E18	6.86 E18	0.0118	30.
87	#1	WELD	0.07	0.082	7.45 E18	6.86 E18	0.0118	60.
88	Cap V	Cor. Mon.	0.68	0.140	7.45 E18	6.86 E18	0.0118	125.
89	PT	A302B-1	0.20	0.19	2.69 E19	2.34 E19	0.0416	105.
90	BEACH	A302B-3	0.20	0.11	2.69 E19	2.34 E19	0.0416	50.
91	#1	WELD	0.57	0.240	2.69 E19	2.34 E19	0.0416	165.
92	Cap R	ASTM A302B Cor.	0.18	0.20	2.69 E19	2.34 E19	0.0416	110.
93	PT	A302B-1	0.20	0.19	9.52 E18	8.44 E18	0.0149	90.
94	BEACH	A302B-3	0.20	0.11	9.52 E18	8.44 E18	0.0149	50.
95	#1	WELD	0.57	0.24	9.52 E18	8.44 E18	0.0149	165.
96	Cap S	Cor. Mon.	0.18	0.20	9.52 E18	8.44 E18	0.0149	95.
97	PT	A302B-1	0.20	0.19	3.50 E18	3.5 E18	0.0062	90.
98	BEACH	A302B-3	0.20	0.11	3.50 E18	(default)	0.0062	50.
99	#1	WELD	0.57	0.24	3.50 E18	3.5 E18	0.0062	110.
100	Cap Q	Cor. Mon.	0.18	0.20	3.50 E18	(default)	0.0062	95.
101	R.E.	A508	0.69	0.05	1.32 E19	1.18 E19	0.0218	60.
102	GINNA	A508	0.69	0.07	1.32 E19	1.18 E19	0.0218	0.
103		WELD	0.56	0.23	1.32 E19	1.18 E19	0.0218	160.
104	Cap R	ASTM A302B Cor.	0.18	0.20	1.32 E19	1.18 E19	0.0218	90.
105	ZION	A533B P.L.	0.49	0.11	2.89 E18	2.83 E18	0.00474	60.
106	#1	A533B P.T.	0.49	0.11	2.89 E18	2.83 E18	0.00474	25.
107	Cap T	WELD	0.57	0.35	2.89 E18	2.83 E18	0.00474	105.
108		HSST-02	0.68	0.14	2.89 E18	2.83 E18	0.00474	66.

TABLE HEDL-1 (Cont'd)

Plant No.	Plant and Capsule	Material	Ni (%)	Cu (%)	Fluence (n/cm^2, E > 1.0 MeV) Randall	Simons	dpa (Simons)	Measured Shift (°F) in Charpy 30 ft-lb Temperature
109	ZION	A533B P.L.	0.49	0.11	8.92 E18	1.0 E19	0.169	85.
110	#1	A533B P.T.	0.49	0.11	8.92 E18	1.0 E19	0.169	60.
111	Cap U	WELD	0.57	0.35	8.92 E18	1.0 E19	0.169	188.
112		Cor. Mon.	0.68	0.14	8.92 E18	1.0 E19	0.169	130.
113	ZION	A533BP.L.	0.53	0.12	2.00 E18	2.86 E18	0.00475	38.
114	#2	A533BP.T.	0.53	0.12	2.00 E18	2.86 E18	0.00475	49.
115	Cap U	WELD	0.55	0.28	2.00 E18	2.86 E18	0.00475	128.
116		Cor. Mon.	0.68	0.14	2.00 E18	2.86 E18	0.00475	50.
117	CONN.	A302B-2	0.20	0.10	1.79 E19	2.19 E19	0.0362	57.
118	YANKEE	A302B-4	0.20	0.12	1.79 E19	2.19 E19	0.0362	67.
119	Cap H	A302B-7	0.20	0.12	1.79 E19	2.19 E19	0.0362	53.
120		ASTM A302B Cor.	0.18	0.20	1.79 E19	2.19 E19	0.0362	127.
121	HADDAM	REPRESENT-	0.20	0.22	1.40 E19	1.4 E19		110.
122	NECK	ATIVE WELD						
	TEST	SURVEILLANCE	0.046	0.22	3.00 E19	3.0 E19	150.	150.
	REACTOR	WELD			(default)			
123	HADDAM	A302B-2	0.20	0.10	2.85 E18	3.05 E18	0.00475	35.
124	NECK	WELD	0.20	0.22	2.85 E18	3.05 E18	0.00475	95.
125	Cap A	Cor. Mon.	0.18	0.20	2.85 E18	3.05 E18	0.00475	85.
126	HADDAM	A302B-2	0.20	0.10	5.54 E18	5.53 E18	0.00838	35.
127	NECK	A302B-4	0.20	0.12	5.54 E18			80.
128	Cap F	A302B-7	0.20	0.12	5.54 E18			50.
129		ASTM A302B Cor.	0.18	0.20	5.54 E18			80.
130	OCONEE	A508 L.	0.75	0.04	9.43 E17	9.92 E17	0.00144	7.
131	#2	A508 T.	0.75	0.04	9.43 E17			0.
132	Cap C	WELD	0.48	0.30	9.43 E17			120.
133		HSST-02	0.64	0.17	9.43 E17			42.
134	INDIAN	A533B-1	0.50	0.18	2.92 E18	3.32 E18	0.0055	89.
135	PT	A553B-3	0.52	0.24	2.92 E18			137.
136	#3	T.	0.52	0.24	2.92 E18			118.
137	Cap T	WELD	1.02	0.34	2.92 E18			143.
138	INDIAN	A302B	1.20	0.25	4.72 E18	4.72 E18	0.00788	145.
139	PT	MODIF.	0.65	0.34	5.89 E18	5.89 E18	0.00944	195.
	#2	WELD				(default)		
	Cap Y							

Default values were not calculated by RL Simons.

P - Plate

T - Transverse Cor Mon - Monitor

L - Longitudinal Tang - Tangential

TABLE 2

RESULTS OF 16 SEPARATE LEAST SQUARES FITS

			Chemistry Function f_1 is Linear Combination of							Exposure Exponent Functional Form	
Computer Run No.	No. of Data Points Used	ID of Points Omitted	Additive Constant	Cu	Cu·Ni Interaction*	Temperature Factor Used	Relative Weight Value Assumed "W"	Standard Deviation (°F)	Exposure Parameter Used	Single Constant	Linear in Log_e (ϕt)
1	136	13,121, 122	X	X	Cu tanh	No	8000	20.32	dpa		X
2	138	13	X	X	Cu tanh	No	8000	20.31	RLS>1		X
3	138	13	X	X	Cu tanh	Yes	8000	20.39	RLS>1		X
4	138	13	X	X	Cu tanh	Yes	8000	20.83	RLS>1	X	
5	139	--	X	X	Cu tanh	Yes	8000	22.36	RLS>1	X	
6	139	--	X	X	$\sqrt{Cu \cdot Ni}$	No	8000	22.33	RLS>1	X	
7	139	--	X	X	$\sqrt{Cu \cdot Ni}$	No	8000	24.26	RLS>1	X	
8	139	--	X	X	Cu·Ni	No	8000	24.31	RLS>1	X	
9	139	--	X	X	Cu·Ni	Yes	2498	25.21	RLS>1	X	
10	139	--	X	X	Cu^2 tanh	Yes	2498	25.19	RLS>1	X	
11	139	--	X	X	Cu·Ni	Yes	9000	24.47	RLS>1	X	
12	139	--	X	X	Cu·Ni	No	2498	26.62	PNR>1	X	
13	136	10,13 68	X	X	Cu·Ni	No	2498	22.51	PNR>1	X	
14	139	--	X	X	Cu·Ni and $Cu \cdot Ni^2$	No	2498	26.11	PNR>1	X	
15	136	10,13 68	X	X	Cu·Ni and $Cu \cdot Ni^2$	No	2498	22.13	PNR>1	X	
16	136	10,13 68	X	X	Cu^2/Ni	No	2498	24.59	PNR>1	X	

RLS - RL Simons
PNR - PN Randall
Entry "tanh" refers to a function of the form: $\tanh\left(x \cdot \frac{Ni}{Cu}\right)$, where x is an adjustable parameter.

OAK RIDGE NATIONAL LABORATORY

(ORNL)

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

F. B. K. Kam
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Objectives

In order to serve as benchmarks, the neutron field at PCA, ORR-PSF, ORR-SDMF, and BSR-HST 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 spectral adjustment 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.
4) BSR-HSST--to study fracture toughness of irradiated pressure vessel materials.

A.1 Pressure Vessel Benchmark Facility for Improvement and Validation of LWR Physics Calculations and Dosimetry (PCA)

Summary

The coupled neutron-gamma calculations for the PCA 12/13 configuration has been completed. The conclusions indicate that in general the revised coupled calculations of Minsart are confirmed, and that a careful analysis of the ^{235}U (n,f) reaction rates show agreement with all reported measurements.

Accomplishments and Status

The coupled PCA 12/13 transport calculations have been completed using the SAILWR cross section library.[1] The thermal group cross sections of the existing 47n-20g coupled set have been modified to correct for the effects of upscattering.

A comparison of the present and earlier calculated saturated activities[2] for threshold monitors indicates good agreement even though the neutron group structure in the present calculation is much coarser (Table 1).

Table 1. Comparison of Present and Earlier Calculated Saturated Activities in the PCA 12/13

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

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

The comparison of the present calculated $^{235}U(n,f)$ reaction rates with measurements indicate absolute agreement to within about 10% at all locations where measurements have so far been reported (Table 2). Earlier calculations did not stress the low energy portion of the neutron spectrum so that a comparison with $^{235}U(n,f)$ was not practical. It is anticipated that similar agreement with the gamma ray measurements will be observed when they become available.

Table 2. Summary of Comparisons of Measured and Calculated $^{235}U(n,f)$ Reaction Rates in the PCA 12/13 in Fissions/Nucleus/Core Neutron

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

[a]Read 2.45×10^{-26} fissions/nucleus/core neutron. MOL fission chamber results in the water, and HEDL SSTR results in the iron. See Ref. 4.

[b]Calculated assuming a cadmium cutoff of 0.414 eV. The corresponding values in water for a cutoff of 0.58 eV are about 10% less (10% more in the Cd ratio). Iron values are little affected.

Table 3 presents the added effect of the photofission reactions to the neutron fissions calculated consistently with the present cross section library, using photofissions cross sections supplied by C. Eisenhauer of NBS. It is seen that the enhancement is small, and the effect is generally in good agreement with the revised calculations of G. Minsart.[3]

Table 3. Photofission Enhancement Effects in the PCA 12/13

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

Finally, comparisons of the gamma-ray fluxes above 6.5 MeV with the neutron fluxes above 0.8 MeV for both the present calculations and the earlier revised calculations of Minsart[3] are shown in Table 4.

Table 4. Comparison of Absolute Neutron and Gamma-Ray fluxes in Units of Particles/cm^2/Core Neutron in the PCA 12/13 for Two Independent Calculations

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

Although the flux comparisons between the two calculations differ at times by factors of up to 1.5 (A6 excepted), the agreement of the gamma ray to neutron flux ratio is excellent for all but detector location A1. The large disagreement in the magnitude of the gamma-ray fluxes at A6 is perhaps related to the disagreement in the Cd ratio between the two calculations there; since the present calculations agree with the measured Cd ratio at A6 (see Table 2) and the Minsart calculations do not, this would favor the gamma fluxes in the present calculations. Until such time as results of gamma-ray measurements become available at this location, however, no conclusions should be drawn.

The conclusions from this study are thus that in general the revised coupled calculations of Minsart are confirmed, and that a careful analysis of the $^{235}U(n,f)$ reaction rates indicates agreement with all existing measurements heretofore reported.

Expected Accomplishments During the Next Reporting Period

Comparisons between calculations and gamma spectrum measurements will be made after the experimental data becomes available.

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

Summary

The final irradiation and temperature distribution data and reactor power time history data for all the LWR-PVS capsule in the ORR-PSF are presented.

Accomplishments and Status

Tables 5-8 represents the final cumulative irradiation and temperature distribution data and reactor power time history data for the LWR-PVS capsules in the ORR-PSF. Minor discrepancies have been noted in previous quarterlies so that all participants in the program are urged to use the data from these tables in their analysis.

Table 5. Cumulative Irradiation and Temperature Distribution Data from April 30-June 23, 1980

Data for PSF Specimen Set SSC-1
Hours of Irradiation Time = 1075.29
Megawatt 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	295.39	3.03
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							
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							
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	9.98	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 6. Cumulative Irradiation and Temperature Distribution Data from May 30–September 25,1981

Data for PSF Specimen Set SSC-2
Hours of Irradiation Time = 2209.87
Megawatt Hours of Irradiation = 64726.56

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE 1	24.99	33.95	2151.26	0.00	0.00	288.40	2.15
TE 2	15.53	8.43	194.00	1960.54	31.34	299.91	2.38
TE 3	20.84	10.01	2168.02	11.00	0.00	291.97	1.82
TE 4	22.46	11.45	2175.95	0.00	0.00	289.37	2.08
TE 5	30.83	705.94	1473.07	0.00	0.00	282.18	2.79
TE 6	33.76	596.93	1579.15	0.00	0.00	282.67	2.62
TE 7	1070.81	1106.04	33.01	0.00	0.00	273.81	1.82
TE 8	24.83	19.48	2096.53	69.01	0.00	289.28	2.63
TE 9	19.02	16.18	1449.54	725.11	0.00	294.88	2.23
TE 10	72.94	2059.88	77.01	0.00	0.00	276.64	2.23
TE 11	40.86	1165.45	1003.52	0.00	0.00	279.29	2.13
TE 12	25.22	10.52	2169.10	5.00	0.00	290.04	2.16
TE 13	22.16	9.46	1623.77	554.46	0.00	293.81	2.34
TE 14	25.10	7.74	2162.01	15.00	0.00	288.92	1.97
TE 15	14.10	12.07	183.13	1979.54	21.00	300.47	1.89
TE 16	24.11	7.65	2178.11	0.00	0.00	290.08	1.37
TE 17	17.16	12.54	1922.44	257.71	0.00	294.39	1.80
TE 18	32.76	1011.70	1165.38	0.00	0.00	294.39	1.80
TE 19	30.72	204.61	1974.48	0.00	0.00	283.38	1.54
TE 20	1475.81	730.05	4.00	0.00	0.00	272.11	1.55

Table 7. Cumulative Irradiation and Temperature Distribution Data From April 30, 1980 to June 23,1982

Data for PSF Specimen Set OT
Hours of Irradiation Time = 14432.03
Megawatt Hours of Irradiation = 427957.42

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE 101	92.85	45.94	14259.81	33.38	0.00	288.39	1.52
TE 102	87.28	31.10	14221.76	91.68	0.00	290.81	1.12
TE 103	86.64	23.48	14321.76	0.07	0.00	289.04	0.90
TE 104	78.94	23.16	14158.29	171.53	0.00	292.03	0.94
TE 105	83.72	32.20	14316.00	0.03	0.00	286.43	0.96
TE 106	79.37	22.03	14330.51	0.00	0.00	289.38	0.90
TE 107	84.88	399.06	13948.05	0.00	0.00	283.54	1.14
TE 108	97.97	38.66	14286.22	9.06	0.00	288.61	1.34
TE 109	99.91	41.55	14281.71	8.78	0.00	288.28	1.40
TE 110	89.29	41.90	14287.82	12.95	0.00	288.62	1.25
TE 111							
TE 112							
TE 113	77.78	23.81	14328.23	0.12	2.00	290.06	1.39
TE 114	107.96	44.21	14279.76	0.00	0.00	287.92	1.39
TE 115							
TE 116	95.44	23.90	14312.58	0.00	0.00	289.91	0.76
TE 117	87.14	27.33	14311.66	5.29	0.50	290.55	0.79
TE 118	90.10	33.97	14307.95	0.00	0.00	287.03	0.85
TE 119	85.50	31.73	14314.75	0.00	0.00	286.91	0.84
TE 120	90.79	257.29	14083.96	0.00	0.00	284.68	1.19

Table 7. (Continued)

Data for PSF Specimen Set 1/4T
Hours of Irradiation Time = 14432.03
Megawatt Hours of Irradiation = 427957.42

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	$T<270$	$270<T<280$	$280<T<296$	$296<T<306$	$306<T$		
TE 201	91.20	35.45	14302.57	2.73	0.00	289.39	1.24
TE 202	91.87	32.85	14307.08	0.17	0.00	288.57	0.75
TE 203	88.24	27.70	14316.06	0.00	0.00	288.59	0.85
TE 204	84.29	26.92	14320.45	0.33	0.00	289.92	0.71
TE 205	84.33	33.69	14312.97	0.00	0.00	287.02	0.81
TE 206	82.64	32.42	14316.95	0.00	0.00	287.39	0.71
TE 207	87.83	119.18	14224.98	0.00	0.00	283.63	0.85
TE 208	91.93	29.31	14309.90	0.83	0.00	288.42	1.09
TE 209	95.20	34.90	14299.92	2.00	0.00	288.71	1.02
TE 210	95.50	46.93	14289.57	0.00	0.00	286.22	0.85
TE 211	101.53	71.41	14259.01	0.00	0.00	283.99	0.77
TE 212	80.25	17.83	14331.79	2.08	0.00	290.62	0.84
TE 213	80.99	19.37	14331.59	0.00	0.00	289.40	0.95
TE 214	95.62	30.72	14305.62	0.00	0.00	290.05	0.94
TE 215	96.41	38.75	14296.76	0.00	0.00	287.23	0.64
TE 216	93.30	29.43	14309.25	0.00	0.00	287.85	0.63
TE 217	88.36	23.54	14320.09	0.00	0.00	289.65	0.77
TE 218	86.10	29.85	14314.03	2.00	0.00	287.29	0.79
TE 219	84.68	25.59	14319.73	2.00	0.00	287.36	0.71
TE 220	84.40	116.65	14230.95	0.00	0.00	286.25	0.99

Table 7. (Continued)

Data for PSF Specimen Set 1/2T
Hours of Irradiation Time = 14432.03
Megawatt Hours of Irradiation = 427957.42

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE 301	91.58	23.60	14277.65	39.18	0.00	298.51	0.83
TE 302	94.75	30.74	14306.53	0.00	0.00	286.62	0.66
TE 303	90.76	26.72	14314.47	0.00	0.00	287.27	0.71
TE 304	82.09	22.19	14325.11	2.58	0.00	291.36	0.66
TE 305	81.92	25.53	14324.47	0.03	0.00	287.59	0.75
TE 306	87.20	29.50	14315.26	0.00	0.00	286.70	0.67
TE 307							
TE 308	93.91	19.04	14319.03	0.00	0.00	288.87	1.05
TE 309	94.52	25.66	14311.83	0.00	0.00	288.09	0.79
TE 310	102.65	50.62	142788.74	0.00	0.00	285.43	0.87
TE 311	99.12	47.86	14285.08	0.00	0.00	285.91	0.89
TE 312	85.30	18.68	14327.81	0.17	0.00	288.44	0.74
TE 313	83.61	19.22	14327.49	1.67	0.00	290.04	0.82
TE 314	97.99	23.64	14310.36	0.00	0.00	288.92	0.90
TE 315	102.94	32.15	14296.91	0.00	0.00	285.27	0.81
TE 316	95.04	17.59	14319.36	0.00	0.00	287.67	0.61
TE 317	86.14	18.78	14327.06	0.00	0.00	290.99	0.72
TE 318	85.43	18.48	14328.04	0.00	0.00	289.40	0.77
TE 319	91.38	31.39	14309.19	0.00	0.00	285.26	0.64
TE 320	86.42	21.48	14324.05	0.00	0.00	287.80	0.98

Table 8. Final Reactor Power Time History for LWR-PVS Capsules in ORR-PSF

Run No.	Inserted	Retracted	Delta-T hours	Delta MWh	Average power	Cummulative hours	Cummulative hours
Start irradiation of SSC No. 1, SPVC, and SVBC							
1	30-Apr-80 13:34	8-May-80 7: 0	184.42	5529.67	29.984	184.42	5529.67
2	8-May-80 16:43	14-May-80 13:30	140.76	4194.60	29.800	325.18	9724.27
3	16-May-80 9:57	21-May-80 2:17	112.33	3365.05	29.957	437.51	13089.32
4	22-May-80 10:49	6-Jun-80 24: 0	370.63	11067.50	29.861	808.14	24156.82
5	12-Jun-80 9:20	23-Jun-80 12:55	267.15	7860.75	29.424	1075.29	32017.57
End irradiation of SSC No. 1							
6	27-Jun-80 18:30	5-Jul-80 3:30	173.68	5107.45	29.407	1248.97	37125.02
7	7-Jul-80 13:55	8-Jul-80 9:40	19.57	573.89	29.325	1268.54	37698.91
8	8-Jul-80 15:18	13-Jul-80 8: 0	111.69	3331.98	29.832	1380.23	41030.89
9	18-Jul-80 17: 0	18-Jul-80 18.32	0.50	4.90	9.800	1380.73	41035.79
10	18-Jul-80 22.50	21-Jul-80 4:26	52.34	1517.90	29.001	1433.07	42553.69
11	22-Jul-80 10: 5	31-Jul-80 7: 0	199.53	6004.28	30.092	1632.60	48557.97
12	31-Jul-80 18:20	12-Aug-80 19: 2	288.37	8757.94	30.370	1920.97	57315.91
13	15-Aug-80 14:48	15-Aug-80 16: 7	1.27	38.25	30.118	1922.24	57354.16
14	21-Aug-80 10:55	26-Aug-80 16: 0	124.69	3608.40	28.939	2046.93	60962.56
15	27-Aug-80 14:30	1-Sep-80 3:29	108.95	3246.64	29.799	2155.88	64209.20
16	3-Sep-80 9:53	9-Sep-80 8: 0	141.55	4268.07	30.152	2297.43	68477.27
17	10-Sep-80 11:22	23-Sep-80 4: 0	302.90	8977.70	29.639	2600.33	77454.97
18	23-Sep-80 13:52	5-Oct-80 21:32	295.23	8843.38	29.954	2895.56	86298.35
19	7-Oct-80 13:46	17-Oct-80 17:50	244.04	7297.95	29.905	3139.60	93596.30
20	21-Oct-80 12:48	29-Oct-80 4: 0	183.13	5429.26	29.647	3322.73	99025.56
21	29-Oct-80 18:47	8-Nov-80 8: 0	228.93	6698.56	29.260	3551.66	105724.12
22	3-Dec-80 14:51	9-Dec-80 0:26	128.68	3730.07	28.987	3680.34	109454.19
23	10-Dec-80 12:54	18-Dec-80 5:15	184.35	5207.41	28.247	3864.69	114661.60
24	18-Dec-80 17:46	30-Dec-80 8: 0	278.23	7758.74	27.886	4142.92	122420.34
25	30-Dec-80 16:11	7-Jan-81 8: 0	183.53	4930.46	26.865	4326.45	127350.80

Table 8. (Continued)

Run No.	Inserted	Retracted	Delta-T hours	Delta MWh	Average power	Cummulative hours	Cummulative hours
26	7-Jan-81 21:55	15-Jan-81 4: 0	173.86	4720.21	27.149	4500.31	132071.01
27	16-Jan-81 11:41	19-Jan-81 20:22	80.47	2425.67	30.144	4580.78	134496.68
28	21-Jan-81 9: 2	22-Jan-81 7:16	21.34	651.57	30.533	4602.12	135148.25
29	22-Jan-81 16:18	2-Feb-81 8: 0	255.09	7759.51	30.419	4857.21	142907.76
30	9-Feb-81 13:35	24-Feb-81 8: 0	351.50	9668.49	27.506	5208.71	152576.25
31	24-Feb-81 15: 0	13-Mar-81 8: 4	398.88	10918.34	27.372	5607.59	163494.59
32	13-Mar-81 8:47	16-Mar-81 3: 0	65.61	1799.07	27.421	5673.20	165293.66
33	19-Mar-81 10:13	30-Mar-81 22:40	276.43	8416.42	30.447	5949.63	173710.08
34	31-Mar-81 11:33	2-Apr-81 4: 0	39.37	1197.23	30.410	5989.00	174907.31
35	2-Apr-81 16:10	19-Apr-81 8: 0	399.84	12111.27	30.290	6388.84	187018.58
36	27-Apr-81 11:12	11-May-81 3:12	325.50	9897.09	30.406	6714.34	196915.67
37	11-May-81 17:24	27-May-81 4: 0	370.60	11241.97	30.335	7084.94	208157.64
		Start irradiation of SSC No. 2					
38	29-May-81 11:39	19-May-81 20:45	9.10	273.44	30.048	7094.04	208431.08
39	1-Jun-81 11:49	9-Jun-81 8:10	187.41	5649.94	30.147	7281.45	214081.02
40	10-Jun-81 8:15	23-Jun-81 4:23	308.13	9352.30	30.352	7589.58	223433.32
41	25-Jun-81 12:20	10-Jul-81 12: 0	359.67	10805.79	30.044	7949.25	234239.11
42	22-Jul-81 13:47	6-Aug-81 6:30	352.57	9552.55	27.094	8301.82	243791.66
43	7-Aug-81 19: 5	20-Aug-81 4: 0	296.93	8019.39	27.008	8598.75	251811.05
44	21-Aug-81 15:17	30-Aug-81 24: 0	224.72	6821.14	30.354	8823.47	258632.20
45	2-Sep-81 19: 1	8-Sep-81 16:52	141.63	4274.86	30.183	8965.10	262907.05
46	11-Sep-81 8:17	25-Sep-81 2: 0	329.71	9977.15	30.260	9294.81	272884.20

Table 8. (Continued)

Run No.	Inserted		Retracted		Delta-T hours	Delta MWh	Average power	Cummulative hours	Cummulative hours
					End irradiation of SSC No. 2				
47	25-Sep-81	23:10	13-Oct-81	3:20	412.05	12439.58	30.189	9706.86	285323.78
48	13-Oct-81	20:30	23-Oct-81	3: 0	221.93	6713.65	30.251	9928.79	292037.43
49	23-Oct-81	13:28	26-Oct-81	20:13	78.63	2378.10	30.244	10007.42	294415.53
50	27-Oct-81	9:41	4-Nov-81	4: 0	185.89	5639.35	30.337	10193.31	300054.88
51	4-Nov-81	16:10	15-Nov-81	8: 0	255.74	7754.49	30.322	10449.05	307809.37
52	24-Nov-81	14:12	12-Dec-81	6: 0	423.79	12835.40	30.287	10872.84	320644.77
53	18-Dec-81	9:47	28-Dec-81	13:20	243.12	7308.76	30.062	11115.96	327953.53
54	31-Dec-81	21:21	6-Jan-82	8:36	130.54	3910.52	29.956	11246.50	331864.05
55	6-Jan-82	14:18	14-Jan-82	3: 0	180.33	5463.26	30.296	11426.83	337327.31
56	21-Jan-82	15:36	1-Feb-82	2:58	251.36	7605.22	30.256	11678.19	344932.53
57	1-Feb-82	16:56	7-Feb-82	8: 0	135.05	4068.56	30.126	11813.24	349001.09
58	12-Feb-82	17:33	18-Feb-82	9: 0	135.38	4061.76	30.003	11948.62	353062.85
59	18-Feb-81	18:59	8-Mar-82	8:20	421.21	12713.18	30.183	12369.83	365776.03
60	9-Mar-82	15:33	25-Mar-82	3: 0	370.01	11185.22	30.230	12739.84	376961.25
61	26-Mar-82	18:55	5-Apr-82	3: 0	223.74	6772.35	30.269	12963.58	383733.60
62	5-Apr-82	18:40	16-Apr-82	15: 5	259.73	7775.02	29.935	13223.31	391508.62
63	29-Apr-82	17:42	24-May-82	3:30	584.92	17610.17	30.107	13808.23	409118.79
64	27-May-82	22:28	22-Jun-82	24: 0	623.80	18838.63	30.200	14432.03	427957.42
					End irradiation of SPVC and SVBC				

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

Summary

The B&W surveillance capsules perturbation experiment is scheduled for irradiation about August 25, 1982 and the shipping of capsules about September 15, 1982.

Accomplishments and Status

All dosimetry capsules for the B&W surveillance capsules perturbation experiment arrived at ORNL in July 1982. The irradiation was rescheduled to August 25, 1982 to accommodate the following changes:

1. a dosimetry capsule was added for insertion back of the void box;
2. thermocouple assemblies were included for insertion into the 1/4 T, 1/2 T, 3/4 T and the two 1/4 T off-set locations; and
3. the original 4/12 configuration was changed to a 4/21.5 configuration.

There were other changes in work tasks because of funding. The two B&W surveillance capsules will be loaded into ORNL's Loop Transfer Cask and shipped to the vendor for disassembly in their hot cells. Art Lowe will provide a purchase order to ORNL for the handling and shipping of the capsules. HEDL will provide a cask to accommodate the MOL and HEDL microtubes and the three void box dosimetry capsules.

Finally the irradiation time was increased from nine days to about twelve days to provide more intensity because of the change in configuration.

Expected Accomplishments in the Next Reporting Period

The irradiation and shipping of the dosimetry capsules and microtubes are scheduled for completion by September 30, 1982.

A.4 Pressure Vessel Benchmark Facility to Study Fracture Toughness of Irradiated Pressure Vessel Materials (BSR-HSST)

Summary

The dosimeters of capsule C of the Fourth HSST irradiation series have been counted.

Accomplishments and Status

The computer program for the statistical analysis of Charpy impact data has been modified and generalized. It is now possible to fit models containing nonlinear functions and products of the input data. The new program was applied to the 61W to 67W series and results were compared with the previously obtained linear fit (see Tables 9 and 10). The range of test conditions is not wide enough to discriminate between different models although the nonlinear models yield better output uncertainties. Results will be presented by R. Berggren at the ASTM Symposium on "Effects of Radiation on Materials," June 28-30, 1982 at Scottsdale, Arizona with F. Stallmann as co-author.

The results of capsules A and B for the Fourth HSST irradiation series were reported in the last quarterly. Capsules C dosimeters have been counted; the analysis is scheduled for the fourth quarter of FY-82.

Expected Accomplishments During the Next Reporting Period

The analysis for capsule C will be completed. Documentation for capsules A, B, and C is scheduled for the first quarter of FY-83.

Table 9. Values of ΔNDT for Different Irradiation Conditions and Copper Content for the 61W-67W Series of Irradiated Weldments

ΔNDT in °F

Specimen Set	Fluence[a] $\phi>1.0$ MeV $*10^{18}$ sec^{-1}	Irradiation[a] Temperature °F	Cu-Content %	Linear Fit Separate[b]	Linear Fit Combined[c]	Nonlinear Fit Separate[b]	Nonlinear Fit Combined[c]
61W	8.27	619°	0.29	131	125	119	123
62W	9.17	563°	0.21	128	138	130	127
63W	7.62	585°	0.30	155	139	148	142
64W	3.92	524	0.35	154	141	146	143
65W	3.55	536	0.22	97	86	99	94
66W	5.05	529	0.42	170	177	154	185
67W	5.03	535	0.27	139	121	151	129

[a]Average value.

[b]Each specimen set is processed separately.

[c]The values for ΔNDT are determined from a fit which includes all specimen and uses the copper content as additional fitting parameter.

Table 10. Values of the Upper Shelf Drop for Different Irradiation Conditions and Copper Content for the 61W-67W Series of Irradiated Weldments

Upper Shelf Drop in ft-lb

Specimen Set	Fluence[a] $\phi>1.0$ MeV $*10^{18}$ sec^{-1}	Irradiation[a] Temperature °F	Cu-Content %	Linear Fit		Nonlinear Fit	
				Separate[b]	Combined[c]	Separate[b]	Combined[c]
61W	8.27	619°	0.29	14	18	13	17
62W	9.17	563°	0.21	27	22	22	23
63W	7.62	585°	0.30	24	20	24	21
64W	3.92	524	0.35	24	22	23	21
65W	3.55	536	0.22	20	19	20	17
66W	5.05	529	0.42	16	23	19	25
67W	5.03	535	0.27	16	21	18	21

[a]Average value.

[b]Each specimen set is processed separately.

[c]The values for ΔNDT are determined from a fit which includes all specimen and uses the copper content as additional fitting parameter.

B. ASTM STANDARDS FOR SURVEILLANCE OF NUCLEAR REACTOR PRESSURE VESSELS

F. B. K. Kam
F. W. Stallmann

Objectives

The primary objective of the LWR Pressure Vessel Surveillance Dosimetry program is to prepare an updated and improved set of dosimetry, damage correlation, and associated reactor analysis ASTM standards to predict the integrated effect of neutron exposure to LWR pressure vessels and support structures.

Accomplishments and Status

The proposed ASTM E706(IID) standard, "Application of Neutron Transport Methods for Reactor Vessel Surveillance," has been approved at the Society level.

The revised versions of the ASTM standards E706(II) and E706(IIA) will be ballotted on the E10.05 level, and E706(IIA) will be ballotted on the E10 level simultaneously.

Expected Accomplishments During the Next Reporting Period

It is anticipated that both E706(II) and E706(IIA) will require minor revisions (hopefully editorial) before ballotting on a higher level.

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3. W. N. McElroy, "LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments and Blind Test," NUREG/CR-1861 HEDL-TME 80-87, R5, Table 5.5.1 (1981).

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16. ABSTRACT *(200 words or less)*

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

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