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RELATING SURVEILLANCE CAPSULE MEASUREMENTS

TO PRESSURE VESSEL DAMAGE*

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DISCLAIMER

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As part of the pressure vessel (PV) materials surveillance program, surveillance capsules including material specimens and neutron flux dosimeters are generally required to monitor changes in the fracture toughness properties of the reactor vessel materials. These capsules are withdrawn sequentially according to a predetermined schedule covering the service life of the vessel, and specimen material changes and dosimeter activation measured. The neutron fluence accumulated by the flux dosimeters is determined from the measured dosimeter activation and known reaction cross section (in practice, the $^{54}Fe(n,p)^{54}Mn$ reaction.) The capsule fluence and material changes are then extrapolated to the pressure vessel using a fluence lead-factor determined from detailed multigroup neutron transport calculations. Typically, in this extrapolation changes in neutron spectrum are neglected. The purpose of this study is twofold; first, to determine the effect of including spectral changes in the extrapolation from capsule to vessel and second, to evaluate the effect of using the latest ENDF/B-V $^{54}Fe(n,p)^{54}Mn$ cross sections in converting dosimeter activation to fluence.

A total lead-factor relating capsule and pressure vessel damage may be defined,

$$L_T = \int_0^\infty \phi^C(E)\sigma^D(E)dE \Big/ \int_0^\infty \phi^V(E)\sigma^D(E)dE \qquad (1)$$

where ϕ^C and ϕ^V are the capsule and vessel fluxes and $\sigma^D(E)$ is the damage cross section of interest. The fluence lead-factor, L_F, generally used to extrapolate from capsule to vessel is based on the flux above 1.0 MeV and is defined,

$$L_F = \int_{1.0\ \mathrm{MeV}}^{\infty} \phi^C(E)dE \Big/ \int_{1.0\ \mathrm{MeV}}^{\infty} \phi^V(E)dE \quad . \qquad (2)$$

L_T may then be expressed as a product of the fluence and spectral lead-factors

$$L_T = L_F \cdot L_S \quad , \qquad (3)$$

where the spectral lead factor is defined,

$$L_S = \tilde{\sigma}^D_C / \tilde{\sigma}^D_V \qquad (4)$$

and the effective damage cross section is given by,

$$\tilde{\sigma}^D_{C(V)} = \int_{0}^{\infty} \phi^{C(V)}(E)\sigma^D(E)dE \Big/ \int_{1.0\ \mathrm{MeV}}^{\infty} \phi^{C(V)}(E)dE \quad . \qquad (5)$$

The pressure vessel damage is then related to the measured activation by,

$$D^{PV} = \frac{\tilde{\sigma}^D_C}{L_F L_S} \; \frac{A_{SAT}}{\tilde{\sigma}^A} \qquad (6)$$

where A_{SAT} is the saturated activity per target atom and $\tilde{\sigma}^A$ is the effective activation cross section defined by replacing σ^D with σ^A in Equation (5).

In order to evaluate both L_S and $\tilde{\sigma}^A$ a one-dimensional ANISN(1) model of the Three Mile Island-2 core, core supports, thermal shield and vessel was constructed. Calculations were performed using both the RSIC-CASK(2) (22-Group, ENDF/B-II) and RSIC-EPR(3) (100-Group, ENDF/B-IV) neutronics Libraries. An S8-P3 angular decomposition was used and the calculations were carried out in the fixed source mode.

The effective damage cross section, $\tilde{\sigma}^D$, was calculated using the damage cross sections developed by Serpan(4) and 100-group ANISN Fluxes. In

Figure 1 $\tilde{\sigma}^D$ is presented for the thermal shield, downcomer and pressure vessel regions. Also presented is the spectral lead-factor, L_S, defined relative to the pressure vessel inner-wall. L_S varies from ~1.22 for an accelerated capsule to ~1.06 for a standard capsule located close to the pressure vessel.

Since $L_S>1$, these results indicate that the ΔTT(change in transition temperature) predicted for the vessel inner wall based on the greater than 1.0 MeV fluence and a fluence lead-factor (i.e., Eq. (6) with $L_S = 1$) will be conservative. However, due to the increase of $\tilde{\sigma}^D$ inside the vessel similar estimates of ΔTT for internal vessel locations are not in general conservative and a spectral lead-factor may be required.

In Table 1 ENDF/B-IV(5) and ENDF/B-V(6) group average cross sections for the ^{54}Fe (n,p)^{54}Mn reaction are presented, together with $\tilde{\sigma}^A$ calculated using a typical 22-Group ANISN capsule spectrum. The ENDF/B-V $\tilde{\sigma}^A$ is low and (using Eq. (6)) will result in an ~2% increase in vessel fluence and damage predictions.

In summary, the effects of spectral changes on the extrapolation of capsule measurements have been evaluated and found to result in a reduction in estimates of PV inner wall fluence and damage. Also, updating the ^{54}Fe(n,p)^{54}Mn activation cross section to ENDF/B-V will increase PV fluence and damage estimates by a few percent.

REFERENCES

1. W.W. Engle, Jr., "ANISN - A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering" K-1693, ORNL (1967).

2. "CASK-40 Group Coupled Neutron and Gamma-Ray Cross Section Data," (1973); available as DLC-23E/ CASK From the Radiation Shielding Information Center (RSIC), ORNL.

3. W.E. Ford III, R.T. Sanford, R.W. Roussin and D. M. Plaster," Modification Number One to the 100n-21γ Cross Section Library," ORNL/TM-5249, Mar. (1976); available as DLC-37D/EPR from RSIC, ORNL.

4. C.Z. Serpan, Jr., "Engineering Damage Cross Sections for Neutron Embrittlement of A302-B Pressure Vessel Steel," Nucl. Eng. Des. 33 (1975) p. 19.

5. ENDF/B-IV, Dosimetry Tape 412 (Rev. 2) Material No. 6417 (Fe-54), Reaction Type 103 (n,p), dated Sept. 1, 1976.

6. ENDF/B-V, Dosimetry Tape 531, Material No. 6430 (Fe-54), Reaction Type 103, (n,p), released Aug. 1979.

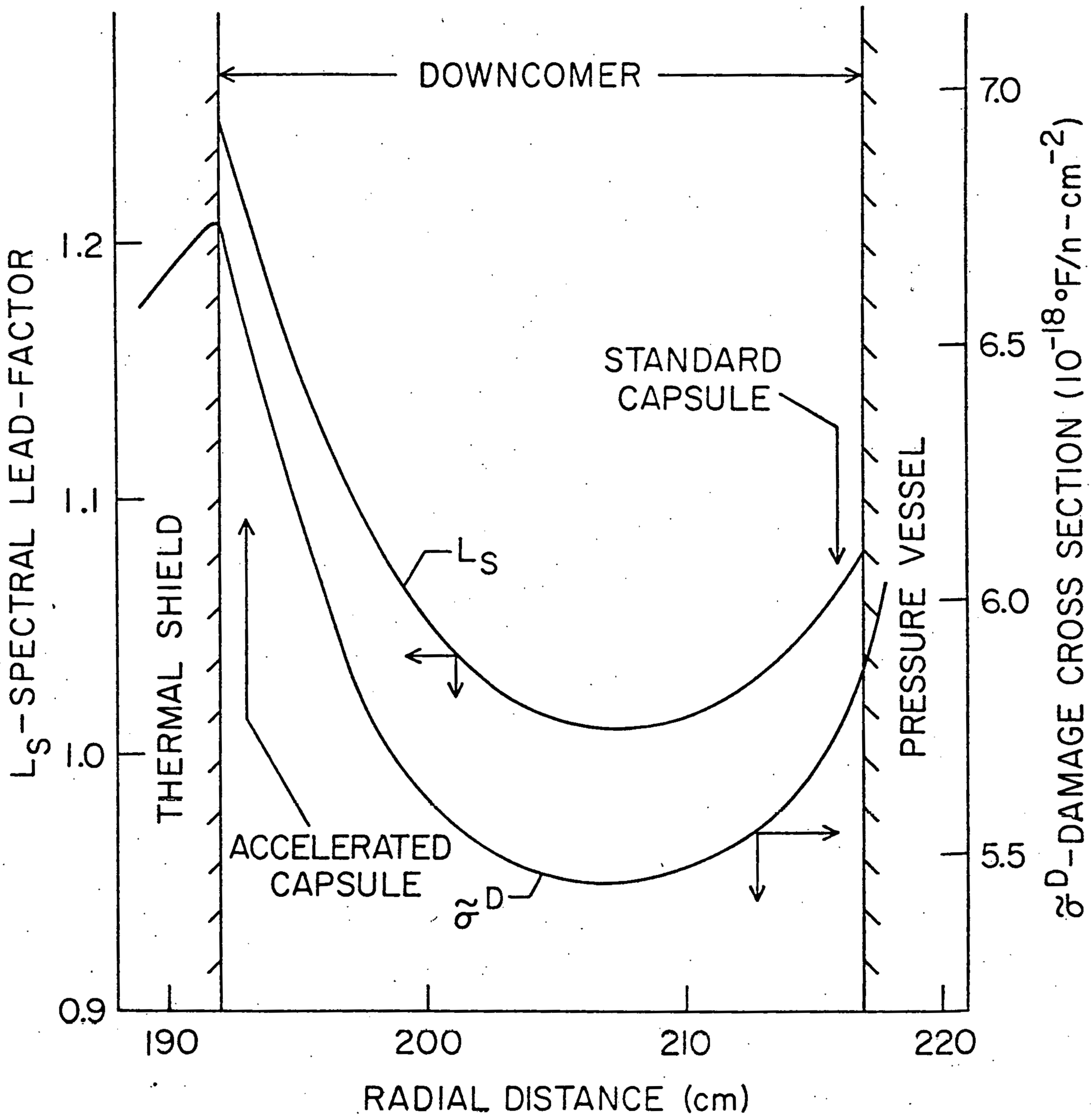

Figure 1   Spectral lead factor and effective damage cross section.

Table 1 Comparison of ENDF/B-IV and ENDF/B-V $^{54}Fe(n,p)^{54}Mn$

Dosimeter Activation Cross Sections†

Group	Lower Energy* Bound (MeV)	Average Cross Section (b)** ENDF/B-IV	ENDF/B-V	Capsule Spectrum	Effective Cross Section (b) ENDF/B-IV	ENDF/B-V
1	12.2	.4272	.4152	.0003	3.886(-4)	3.777(-4)
2	10.0	.5371	.4728	.0012	1.673(-3)	1.473(-3)
3	8.18	.5833	.4772	.0032	5.010(-3)	4.100(-3)
4	6.36	.5720	.4714	.0089	1.351(-2)	1.113(-2)
5	4.96	.4745	.4321	.0170	2.134(-2)	1.944(-2)
6	4.06	.3262	.3276	.0172	1.485(-2)	1.492(-2)
7	3.01	.2082	.2196	.0308	1.696(-2)	1.789(-2)
8	2.46	.09658	.1082	.0359	9.178(-3)	1.028(-2)
9	2.35	.05234	.05613	.0113	1.559(-3)	1.672(-3)
10	1.83	.02229	.02962	.0575	3.393(-3)	4.508(-3)
11	1.11	.00116	.00297	.1556	4.764(-4)	1.225(-3)
12	.55	0.0	7.139(-5)	.2618	0.0	4.946(-5)
13	.11	0.0	0.0	.3993	0.0	0.0
					$\tilde{\sigma}^A$ = 8.835(-2)	8.706(-2)

† 7.14(-5) is read 7.14×10^{-5}.
* Upper energy of Group 1 is 15.0 MeV and the group structure is the same as in CASK.
** Cross sections collapsed using a watt fission spectrum (a=0.988 MeV, b=2.249 MeV^{-1}) joined to a 1/E spectrum at 67.38 KeV.