Request from Radiation Damage Evaluation in Materials

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Radiation transport calculations in a PWR using cross-section data sets based on JENDL3.2 showed that the calculated neutron fluence agreed well with the dosimeter measurements and that the fast neutron flux and dpa rate differed within 10% from to those calculated using ENDF/B-IV and ENDF/B-VI based data sets. Calculations of helium generation in structural materials in the PWR using ENDF/B-VI showed that the dominant source of helium is the (n, α) reaction of ^{59}Ni and that the calculated helium content agreed with the measurements. For accurate estimation of radiation field from a material viewpoint, it is desirable to construct proper cross-section libraries, which have a proper energy group structure and contain sufficient elements including ^{59}Ni as an indispensable element.

1. Introduction

Accurate characterization of radiation field is essential for precise life prediction of structural materials and understanding of material degradation mechanisms in nuclear reactors. Nuclear cross-section data (nuclear data) are used for evaluating fluxes and spectra of neutron and gamma ray in near-core components such as reactor vessels, for measuring radiation fields using dosimeters and for modeling microstructural evolution based on radiation damage parameters such as cascade formation and gas generation.

Material property changes under radiation field are determined by parameters such as radiation fluence, flux, temperature, stress and material composition. Fast neutron fluence with the energy higher than 1MeV or 0.1MeV is widely used as a traditional exposure for prediction of material properties in various reactors. Displacement per atom (dpa) is also used as an exposure parameter. Procedures of dpa calculation and dpa cross-section of iron under neutron irradiation have been standardized in ASTM E693. This standard has been derived from ENDF/B and widely used in nuclear industry. The dpa induced by gamma rays is evaluated for structural materials in which the gamma ray flux is expected much higher than neutron flux.[1)] Nuclear transmutation reactions cause a gradual change in material composition. Although the change in material composition is negligible for major metallic elements in structural materials, the production of helium and hydrogen through (n, α) and (n,p) reactions is known to have a strong effect on material property change such as swelling and creep. Recently it has been demonstrated that the helium generation has a detrimental influence on weldability of irradiated materials.[2)] The weldability of irradiated stainless steels is degraded by the existence of 0.1 - 1 appm helium. The accurate estimation of helium generation is important to assess applicability of welding to irradiated structural materials. Recent experiments showed that swelling initiation is sensitive to material temperature during irradiation and that an increase of 10 ºC resulted in a larger swelling.[3)] The temperature of structural materials during reactor operation is estimated using heat transfer calculations considering gamma heating and coolant flow distribution. The accuracy of gamma heating is a key factor for the accuracy of temperature estimation.

As mentioned above, the accuracy of parameters that determined material behaviors under reactor radiation environment depends on accuracy of nuclear data and calculation modeling. In this paper, results of radiation transport calculations and dpa calculations inside the reactor vessel of a two-loop pressurized water reactor (PWR) using JNEDL data are presented and compared with those using ENDF/B data. Estimation of helium generation in core materials of the PWR is also presented. Finally requests for domestic nuclear data are summarized from the viewpoint of material evaluation under radiation field.

2. Neutron flux and dpa calculated using JENDL3.2

The transport calculations in a two-loop PWR (1456MWth) were carried out using DORT code with three cross-section libraries (JSSTDL[4)], BUGLE-96[5)] and JSD-100[6)]), TORT code with BUGLE-96 and MCNP with JENDL3.2. The JSSTDL library is generated from JENDL3.2 and has a 100-neutron and 40-gamma group structure. The JSD-100 data set was generated from the JSD-100 library (100-neutron and 40-gamma group, based on ENDF/B-IV) and has a 21-neutron and 13-gamma group structure collapsed using the spectrum calculated with the one-dimensional ANISN code for the PWR. This data set has been used for fluence evaluation in PWRs in Japan for almost two decades. In this paper it is simply designated as JSD-100. The BUGLE-96 is generated from ENDF/B-VI and has a 47-neutron and 20-gamma group structure. The one-eighth horizontal geometry model is shown in Fig.1. The detailed calculation procedure was described elsewhere.[7)] Figure 2 shows the spectra of neutrons and gamma rays at the inner surface of the reactor vessel. The spectra calculated with various code and libraries are well coincident with each other. Table 1 shows the C/M ratios for the surveillance dosimeters (Fe, Ni, Cu, ^{238}U and ^{239}Np) installed at the outer surface of the thermal shield. The three libraries used in the calculation gave almost the same average C/M ratio (0.98 – 1.05) for the surveillance dosimeters, indicating that these libraries gave sufficiently precise estimation of neutron flux at almost the same level. The fast neutron fluxes at the surveillance position and the reactor vessel calculated with JSSTDL agreed within 10% with those calculated with BUGLE-96 and JSD-100.

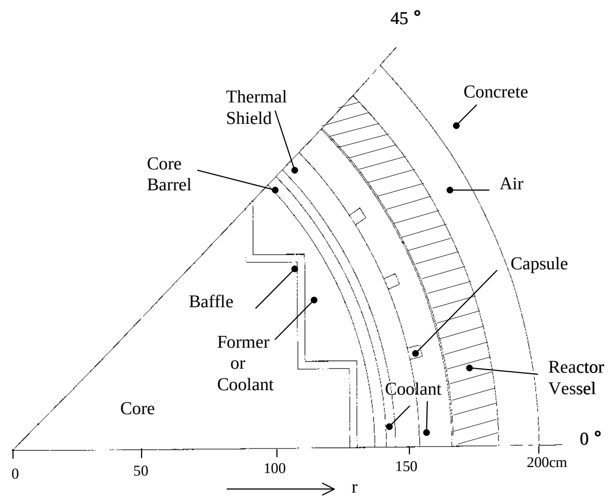

Fig.1 One-eighth horizontal geometry of a two-loop PWR

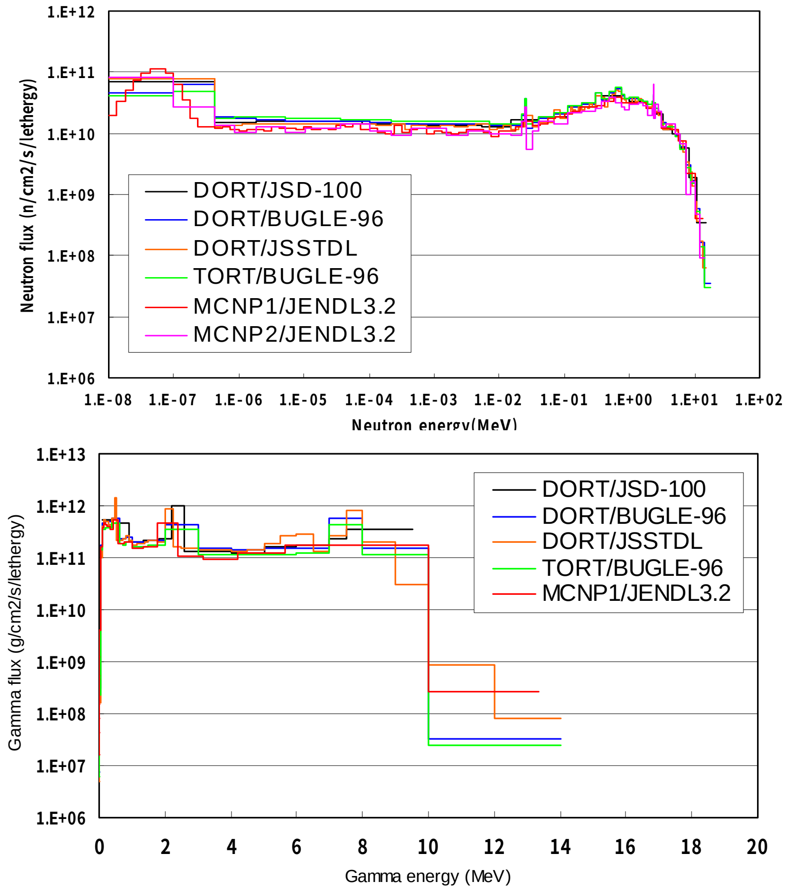

Fig.2 Spectra of neutrons (upper) and gamma rays (lower) at the inner surface of reactor vessel calculated using various codes and libraries.

Table 1 C/M ratios for surveillance dosimetry

Code and data set		^{54}Fe (*n, p*) ^{54}Mn	^{58}Ni (*n, p*) ^{58}Co	^{63}Cu (*n,α*) ^{60}Co	^{238}U (*n, f*) ^{137}Cs	^{237}Np (*n, f*) ^{137}Cs	Average (Standard Deviation)
TORT	BUGLE-96	0.94	1.12	0.91	0.96	1.12	1.01 (0.10)
DORT	JSD-100	0.90	1.04	1.19	0.92	1.19	1.05 (0.14)
	BUGLE-96	0.94	1.13	0.95	0.96	1.14	1.03 (0.10)
	JSSTDL	0.91	1.09	0.96	0.92	1.04	0.98 (0.08)
MCNP1	JENDL3.2	0.90	1.05	0.96	0.76	0.80	0.89 (0.12)
MCNP2	JENDL3.2	0.88	1.04	0.96	0.79	0.83	0.90 (0.10)

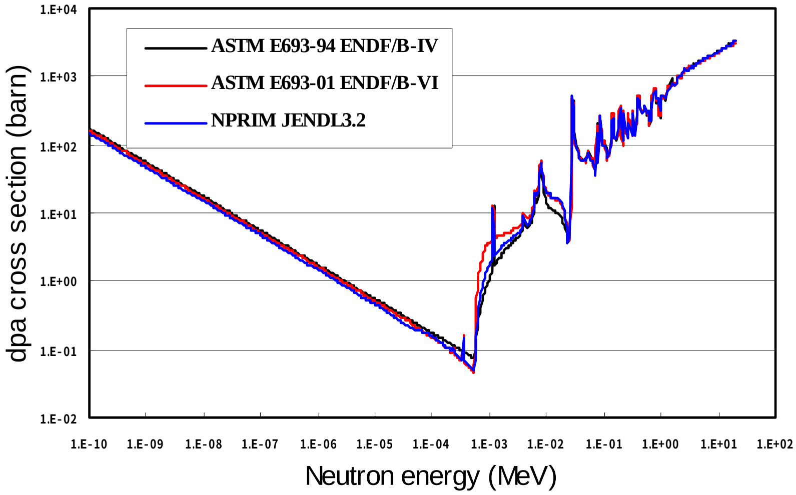

Fig. 3 Dpa cross-sections in iron

The neutron induced dpa in structural materials inside the reactor vessel was calculated using dpa cross-sections in iron derived from the three nuclear data. The dpa cross-section from JENDL3.2 was calculated using NPRIM code.[8)] The dpa cross-sections tabulated in the ASTM E693 standard (E693-94 and E693-01) were used for those from ENDF/B-IV and ENDF/B-VI, respectively. Fig. 3 compares the energy dependence of dpa cross-sections in iron. A small difference was observed in the energies from keV to 10keV. The gamma ray induced dpa cross-section in iron has not been standardized while several dpa cross-sections were proposed.[1, 9, 10)] In this study a gamma ray induced dpa cross-section was calculated taking into account Compton scattering, photoelectric effect and pair production, using the McKinley-Feshbach approximation for the electron and positron displacement cross-section, the NRT model[11)] for the displacement function and 40eV for the displacement threshold energy. Both neutron-induced dpa and gamma-induced dpa at the surveillance position and the reactor vessel based on JEDNL3.2 agreed within 15% with those based on ENDF/B-IV and ENDF/B-VI.

The results described above indicate that JENDL3.2 gives sufficiently accurate fluxes of neutrons and gamma rays and dpa inside the reactor vessel of PWRs for engineering purpose. JENDL3.2 gives almost the same estimations of neutron fluence and dpa as ENDF/B currently used for PWR application.

3. Helium generation

The amount of helium generation through (n, α) reactions in stainless steel components in the PWR was estimated using cross-section data in ENDF/B-VI and neutron fluxes calculated with BUGLE-96. The ENDF/B-VI alone includes cross-section data of ^{59}Ni isotope. Fig.4 shows the relative contribution of the isotopes to the total helium generation at a baffle plate. The composition of the plate was assumed Fe - 18Cr - 8Ni - 0.06N - 0.0009B in wt %. At the beginning of irradiation up to ten years the ^{10}B(n,α) reaction is the main source of helium. After ten years the ^{58}Ni (n, p) ^{59}Ni (n, α) ^{56}Fe reaction becomes dominant and its contribution reaches to 90% of the total helium generation. Fig.5 shows the comparison of calculated helium generation and measured helium content in a thimble tube made of type 316 stainless steel. The detailed data was described elsewhere.[12)] The thimble tube was installed in a fuel bundle for 13 effective full power years. The calculated values were in good agreement with the measurements for the sample B, C, D and E. However no good agreement was observed for the samples A, G and H, which were located near the thermal flux peak positions outside the active fuel length. Precise estimation of helium generation needs accurate thermal neutron flux and cross-sections of ^{59}Ni isotope.

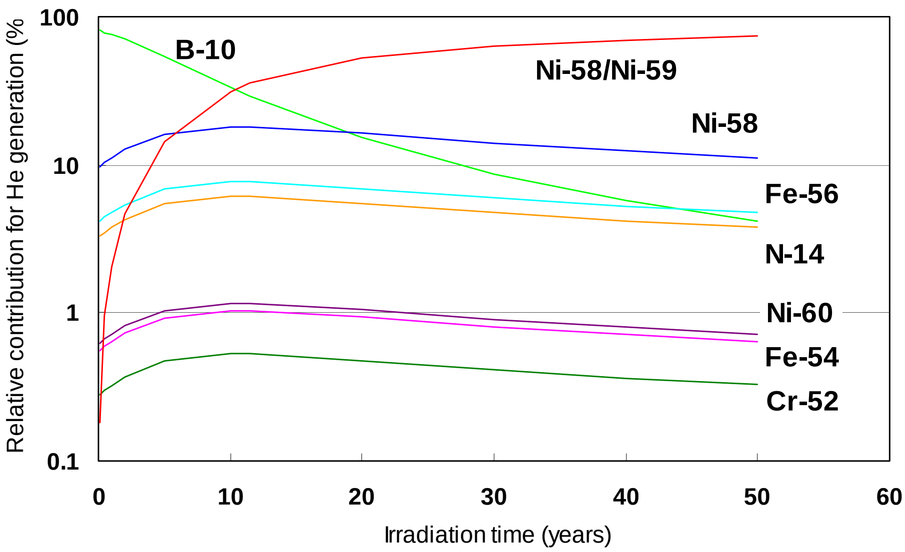

Fig.4 Relative contribution of isotopes to the total helium generation in in-core structural material.

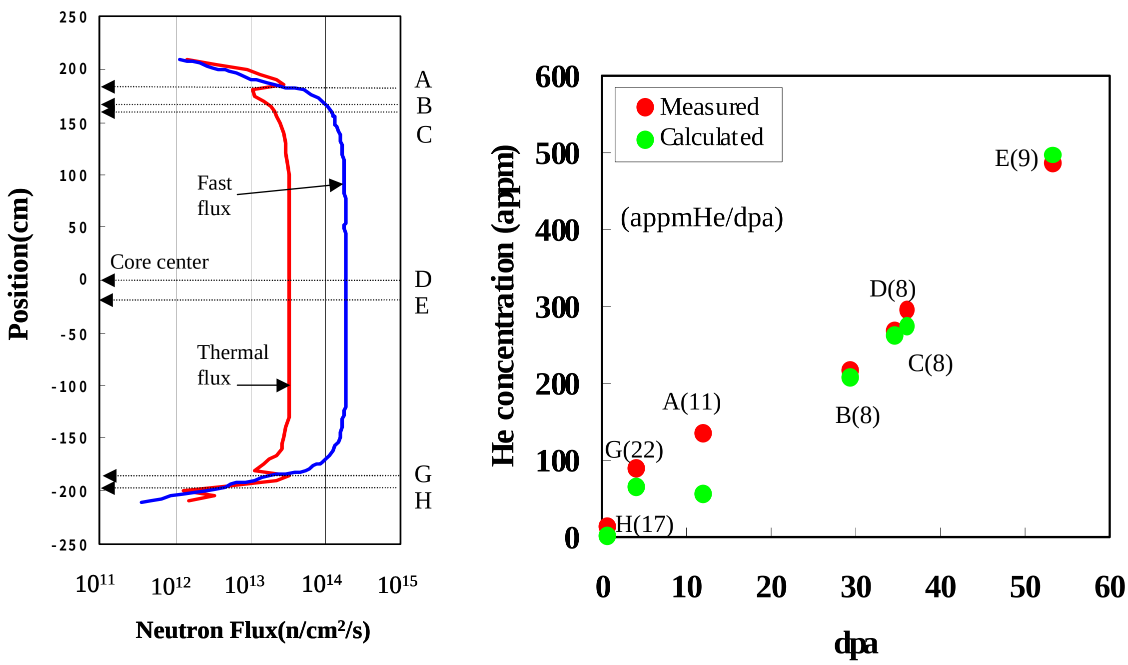

Fig.5 Comparison of measured helium content and calculated helium generation in in-core structural material (right) and specimen positions along the core height (left).

4. Summary and requests to JENDL

The present calculations using JSSTDL and JENDL3.2 demonstrated that JENDL gives sufficiently accurate estimation of neutron flux and dpa rate in PWR. However JENDL is not suitable for calculations of helium generation because of the lack of ^{59}Ni cross-section data. Requests to JENDL can be summarized as follows for accurate estimation of radiation field from a material viewpoint. First it is desirable to construct proper cross-section libraries that have an energy group structure suitable to material evaluation. The thermal region should contain several groups to improve the accuracy of thermal flux. The libraries examined in this paper contain one group (in JSSTDL and JSD-100) or two groups (in BUGLE-96). This probably is one of the reasons for a large difference in thermal flux between the three libraries. For the fast region the group division at 1 and 0.1 MeV is desirable. The division in JSSTDL and BUGLE-96 is 0.111 MeV and in JSD-100 is 0.1 MeV. Although the difference in fast flux between divisions at 0.1 MeV and 0.111 MeV may be small, factors causing data scattering in database analyses should be minimized. Secondly for evaluating generation of helium and hydrogen the cross-section data of ^{59}Ni must be included as an indispensable element. Finally standard dpa cross-sections in iron and stainless steels derived from JENDL should be prepared, as the ASTM E693 has been standardized from ENDF/B.

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