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A Preliminary Shielding Design of 150MWe Liquid Metal Reactor

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A preliminary shielding design of the KALIMER (Korea Advanced Liquid Metal Reactor) was established by two-dimensional discrete ordinate radiation transport analyses. The shielding design has tentatively adopted to use two limits on the fast neutron fluence and the DPA (Displacement Per Atom) simultaneously as a base for the exposure limit for neutron damages to reactor structures. In addition, activities in the PSDRS air effluent and the IHX secondary sodium were also examined.

The DORT two-dimensional transport code was used to evaluate the KALIMER shielding design. The reactor system was represented by four axial zones, each of which was modeled in the R-Z geometry. The KAFAX-F22 library was used in the analyses, which was generated from the JEF-2.2 of OECD/NEA for LMR applications by KAERI (Korea Atomic Energy Research Institute), and consists of 80 neutron and 24 gamma energy groups.

The performance of the shielding design is compared against the shielding design criteria. The results indicate that the support barrel, upper grid plate, and other reactor structures meet both the maximum neutron fluence and DPA limits established in the shielding design criteria. Activities of the air effluent in the PSDRS were also evaluated and are shown to satisfy the effluent concentration limits in 10 CFR Part 20.

It is found that maximum DPAs show larger margins than the case when the neutron fluence limit is used as a design criterion. Therefore, the use of DPA as a shielding design criterion allows for a more flexibility in the LMR shielding design.

KEYWORDS: Fast reactor shielding, fast neutron fluence, displacement per atom, neutron irradiation damage, secondary radiation source generation, DORT, discrete ordinate method, design evaluation, residual total elongation

I. Introduction

The KALIMER[1] (Korea Advanced Liquid Metal Reactor) is a modular liquid metal reactor in the conceptual design phase, whose electric output is 150MWe. The primary system of KALIMER is a pool type, and four pairs of EM pumps and IHXs (Intermediate Heat Exchangers) are arranged between the support barrel and the reactor vessel. The reactor core is configured to be radially homogeneous, and employs the IVS (In Vessel Storage) in the outermost region of the reactor. The removable B_4C shield assembly and steel shields are located inside and outside of the IVS respectively.

The KALIMER has a PSDRS (Passive Safety Decay Heat Removal System) for cooling of the containment vessel by natural airflow. The KALIMER reactor system layout and core configurations are shown in **Fig. 1**.

Most of the principal system components such as EM pump and IHX are located inside the reactor vessel contrary to a conventional PWR, where these components are placed outside of the pressure vessel. The structures of the reactor are confronted by the condition of the higher neutron flux and energy. Accordingly, the irradiation condition of the structure in a fast reactor is more severe than that in a thermal reactor.

The use of sodium as a primary and secondary coolant generates the secondary radiation source by neutron activation. This radiation source results in an increase of radiation exposure for the worker in the radiation zones.

The shielding design was concentrated around the core to prevent adjacent principal structures from the neutron damage and protect the worker from radiation exposure by reduction of the secondary radiation source generation.

Neutron exposure limits for reactor structures are derived on the basis of material ductility decrease with irradiation. Neutron damage limits that ensure 10% residual total elongation (RTE) at end-of-life for load bearing structures and 5% for non-load bearing structures have been adopted as the design criteria. The shielding design has tentatively adopted to use two limits on the neutron fluence and DPA (Displacement Per Atom) simultaneously as a base for exposure limit for neutron damages to reactor structures. The fast neutron fluence is the value that has been used for the neutron damage estimation at the pressure vessel in a thermal reactor. While, the DPA is the value that gives more explicit account for the neutron damage, and includes the neutron spectrum effect.

In addition, activities in the PSDRS air effluent and the IHX secondary sodium were also examined to estimate the secondary radiation source production.

The shielding design criteria used in this analysis are given in **Table 1**.

II. Shielding Design

The KALIMER shielding design employs three types of shields: inner fixed shield, radial PSDRS shields, and IHX shields according to their locations. The radial PSDRS shields are three B_4C cylinders located at the outside of the support barrel at core level with thickness of 15cm and are to reduce the activation of the PSDRS air effluent and protect the reactor

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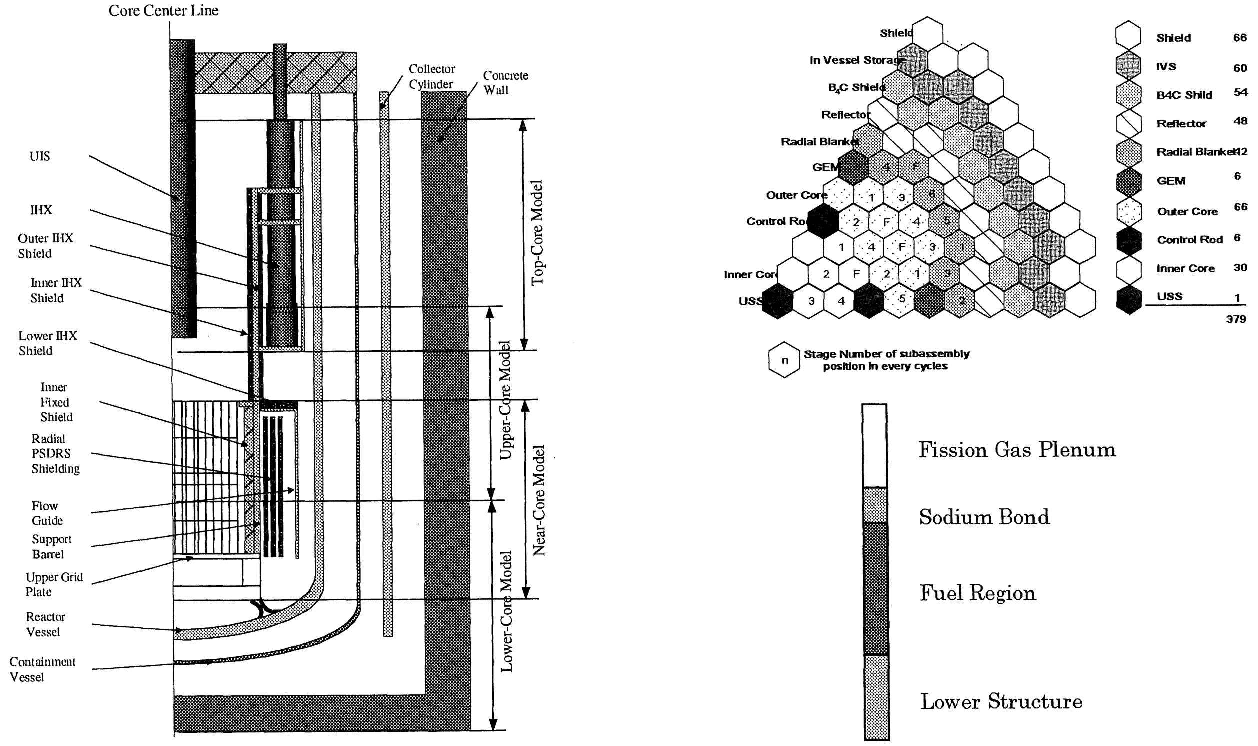

Fig. 1 Reactor System Diagram and 1/6 Core Configuration

Table 1 The Tentative Shielding Design Criteria

Fast neutron fluence (E > 0.1MeV) for structural material	< 5.0 ×10^{21} n/cm^2
DPA for load bearing structures	
For SS316	< 4.1
For SS304	< 2.4
DPA for non-load bearing structures	
For SS316	< 8.2
For SS304	< 4.8
Secondary sodium activity in IHX	< 4x10^4 Bq/cc
PSDRS air effluent activity	< 2x10^6 μCi/cc[(a)]

[(a)] ^{41}Ar activity[(6)]

vessel and containment vessel from neutron irradiation. The inner fixed shield is 8cm thick, and is located at the inside wall of the support barrel to protect that from neutron exposure.

The inner and outer IHX shields were mounted on the inner and outer walls of the support barrel above the elevation of the core and each has a thickness of 8cm and 10cm. The lower IHX shield was located above the flow guide with thickness of 15cm. The role of the IHX shields is to reduce the secondary radiation source generation due to activation of the sodium in the IHX. All shields were composed of natural B_4C powder with the density of 2.27 g/cc with 0.9 packing factor. Usually, a shield is used in the form of a solid plate or can containing the absorber in a ball or rod type. In this work, it was assumed that all shields were made of B_4C powder only for the preliminary design purpose.

III. Analytical Methods

The shielding analyses were performed using the DORT[(2)] two-dimensional discrete ordinate particle transport theory code and the KAFAX-F22[(3)] neutron-gamma coupled library, which was generated from JEF2.2 of OECD/NEA for LMR application by KAERI. The detailed descriptions of the analytical model are as follows:

1. Cross-section Library

KAFAX-F22 library consists of 80 neutron and 24 gamma groups in the MATXS format. In order to use this library with the DORT code, TRANSX[(4)] code was used to convert the library to the ANISN library format and generate the macroscopic cross-sections for each R-Z DORT model.

KAFAX library provides not only basic cross-sections for transport codes, but also extra cross-sections for editing the heating rate, DPA, and activity. The damage energy production cross-section was used to calculate the DPA for neutron damage estimation in this work.

2. Fission Neutron Sources

The spatial fission neutron sources were obtained from the DORT eigen-value calculation modeling only the reactor core in order to speed up the convergence and reduce the computation time. The fission neutron spectrum was taken from those in the outer core region.

3. DORT Models

Two-dimensional R-Z DORT models were used in these analyses. This R-Z representation of the core could envelop most of the reactor components with consideration of the azimuthally homogeneous core configuration.

The geometrical model extends axially from the concrete floor below containment vessel to the top of the IHX, and radially from the core centerline to the concrete wall. Because of a limitation of memory space, the reactor system was divided into four overlapping axial regions: near, lower, upper, and top core model. In DORT calculations, the top-core model was coupled with a boundary source to the upper-core model because of the absence of the fission neutron source in the region. The axial elevations of each model are shown in Fig. 1.

An S_8 quadrature and P_3 scattering expansion were specified in each model input. The radial and axial dimensions of each model were between 4.5m and 6m and mesh sizes were between 1cm and 4cm. Considering that the neutron mean free path is longer than that of a thermal reactor, the mesh sizes seems to be sufficiently small.

Due to the limitation of the R-Z geometrical representation, some of structures configured by the azimuthally heterogeneous form were either omitted from the model or represented as azimuthally homogeneous regions to be conservative. The EM pumps were excluded from the model, because the values in the IHX could represent those for the EM pump. Four IHXs were modeled as an azimuthally homogeneous region around the support barrel with the same width.

In order to examine the effectiveness of the shielding on reactor components, DORT calculation for two different shielding configurations was made as followings:

Case 1: Model with inner fixed shield, radial PSDRS shield, and IHX shield,

Case 2: Model having no shield.

4. DPA Evaluation Method

DPA was calculated from the damage cross section as following,

$$\text{DPA} = T \cdot \sum_{g} \sigma_{d,g} \phi_g \tag{1}$$

where T is the operation time, $\sigma_{d,g}$ the damage cross section (barn), and σ_g the neutron group flux (n/cm^2-sec).

The damage cross section was obtained by using the TRANSX code based on the following formula[(5)],

$$\sigma_{d.g} = \frac{\chi}{2E_d} \sigma_{DAME.g} \tag{2}$$

where χ is the displacement efficiency (0.8), E_d the displacement energy (40 eV for Fe), and $\sigma_{DAME,g}$ the damage energy production cross section (eV-barn) included in the KAFAX-F22. The damage cross section of ^{56}Fe and neutron flux at the support barrel are depicted in **Fig. 2**.

IV. Results

The reactor models have been analyzed by using the DORT two-dimensional transport code. The 30-year design lifetime and 0.85 capacity factor were considered in this study. The calculation uncertainties were not considered in this preliminary analysis.

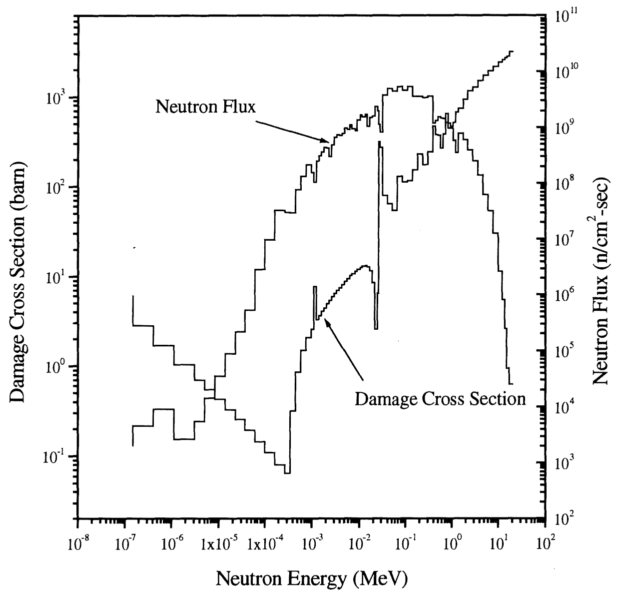

Fig. 2 Damage Cross Section for ^{56}Fe and Neutron Flux

The fast neutron fluence (E > 0.1MeV), DPA, and activities of air in PSDRS and of sodium in the IHX were compared against the shielding design criteria and their results are summarized below:

1. Neutron Damage Estimation

The key components of concern for neutron irradiation damage are the support barrel and upper grid plate, which are most close components to the reactor core. In **Table2**, it is shown that the maximum neutron fluence at the support barrel inner surface without a shield is 2.02x10^{21} neutrons/cm^2. This value marginally satisfies the design limit. The placement of the inner fixed shield resulted in a reduction of the maximum neutron fluence in the support barrel to 2.40x10^{20} neutrons/cm^2, which meets the neutron fluence limit with sufficient margin.

The maximum neutron fluence in the upper grid plate is unchanged because no shielding is provided for the upper grid protection as shown in Fig. 1. But the lower structure below active fuel provides shielding to the upper grid plate.

The maximum DPA in the support barrel was reduced from 7.48x10^{-1} to 9.86x10^{-2} with the shieldings. For the upper grid plate, the maximum DPA is 3.9x10^{-2}. It is shown that the maximum DPA in the support barrel and the upper grid plate are well below the design limits in both unshielded and shielded cases. The numbers in the parenthesis represent the ratios between calculated results and design limits. Comparisons of these ratios indicate that DPA results in larger design margins compared to those of the neutron fluence.

The neutron fluence and DPA in other reactor components are held below the design limit. The results of the neutron damage calculation for the principal reactor components are summarized in Table 2 and the contour plots of fast neutron fluence in each model are depicted in **Fig. 3** through **Fig. 6.**

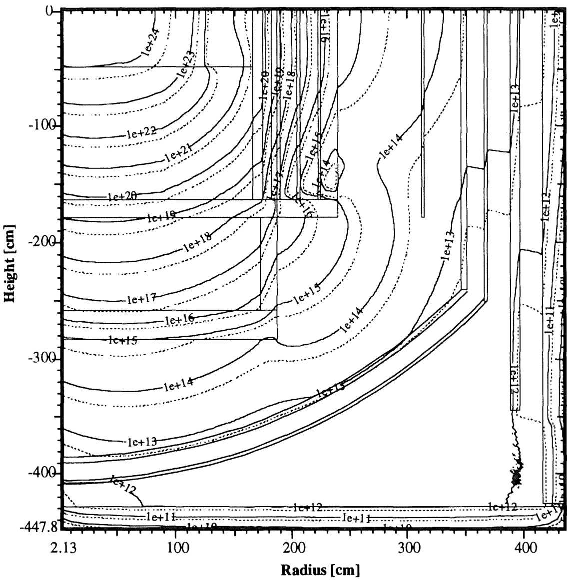

Fig. 3 Fast Neutron Fluence with Shields for Lower-core Model

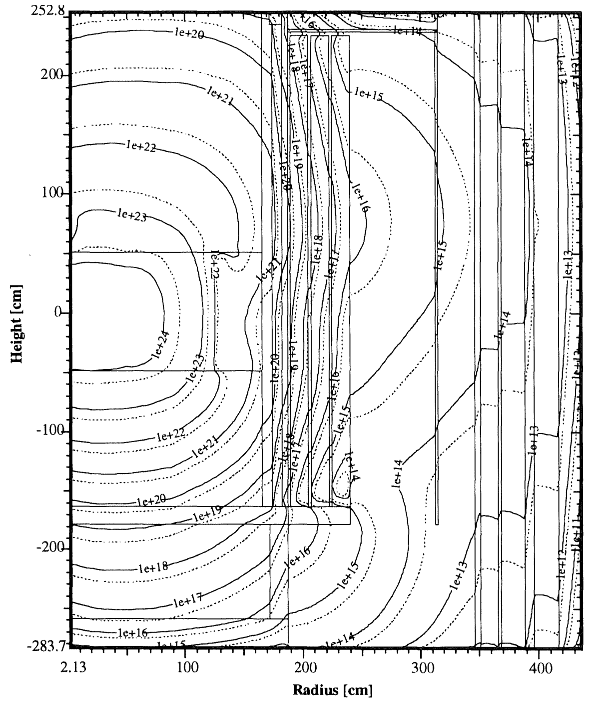

Fig. 4 Fast Neutron Fluence with Shields for Near-core Model

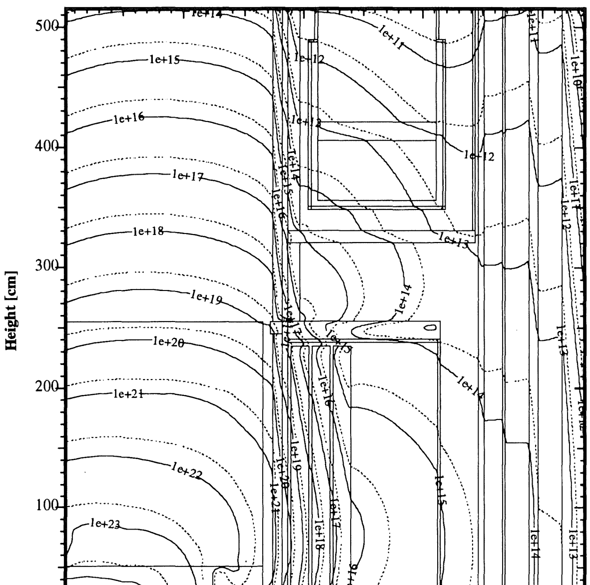

Fig. 5 Fast Neutron Fluence with Shields for Upper-core Model

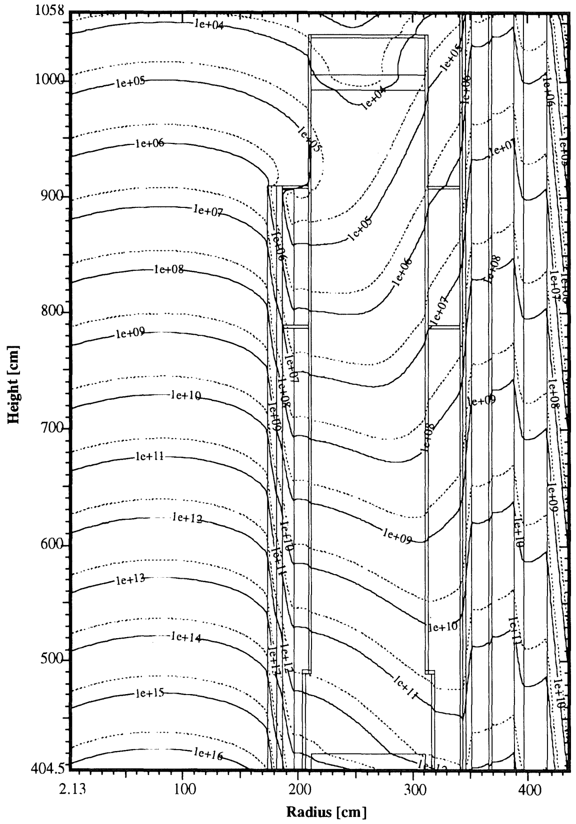

Fig. 6 Fast Neutron Fluence with Shields for Top-core Model

Table 2 The calculated maximum fast neutron fluence and DPA for each structure

(Unit: n/cm²)

Components	Material.		Design Limit	DORT results	
				Case 1	Case 2
Support barrel	SS316	Fluence	< 5.0E+21	2.02E+21(4.04E-01)*	2.40E+20(4.80E-02)
		DPA	< 4.1	7.48E-01(1.82E-01)	9.86E-02(2.40E-02)
Upper grid plate	SS304	Fluence	< 5.0E+21	7.45E+19(1.49E-02)	7.45E+19(1.49E-02)
		DPA	< 2.4	3.90E-02(1.63E-02)	3.90E-02(1.63E-02)
Reactor vessel	SS316	Fluence	< 5.0E+21	6.64E+17(1.33E-04)	4.34E+14(8.68E-08)
		DPA	< 4.1	5.61E-04(1.37E-04)	1.83E-07(4.46E-08)
Containment vessel	Cr-Mo Alloy	Fluence	< 5.0E+21	2.91E+17	2.14E+14
		DPA	-	2.06E-04	8.55E-08

* The numbers in the parenthesis are the ratios between calculated results and design limits

Table 3 Secondary radiation source generation in the IHX and the PSDRS

	Reaction	Design Limit	DORT results	
			Case 1	Case 2
Secondary sodium activities (Bq/cc)	^{23}Na (n,2n) ^{22}Na	-	9.16E-03	6.71E-04
	^{23}Na (n,γ) ^{24}Na	< 4.0E+04	2.06E+07	1.76E+02
	^{23}Na (n,p) ^{23}Ne	-	2.16E+00	1.59E-01
PSDRS air effluent activities (μCi/cc)	^{40}Ar(n,γ)^{41}Ar	< 2.0E-06	3.04E-05	9.05E-09
	^{14}N(n,p)^{14}C	-	5.19E-10	1.51E-13
	^{16}O(n,p)^{16}N	-	1.17E-07	6.06E-11

2. Secondary Radiation Source Generation

Neutrons from the reactor core cause activation of the secondary sodium in the IHX, which becomes a source of radiation in the IHX piping in other areas of the plant. Activities of ^{22}Na, ^{24}Na, ^{23}Ne resulting from sodium activation were calculated for the cases with and without shields and are summarized in **Table 3**. The major contributor among these isotopes was found to be ^{24}Na generated by radiative capture reaction with ^{23}Na. Other isotopes were negligible compared with ^{24}Na activity.

The maximum activity of ^{24}Na is 2.06×10^{7} Bq/cc when there is no shield. This value exceeds the design limit on secondary sodium activity, but the activity drops to 1.76×10^{2} Bq/cc when all shields are in place.

The activities of ^{41}Ar, ^{14}C, and ^{16}N in the PSDRS air effluent were calculated and summarized in Table 3. ^{41}Ar was the dominant radiation source and its maximum activity was found to be 3.04×10^{-5} μCi/cc without a shield. The activity was reduced to 9.05×10^{-9} μCi/cc when fully shielded. The air effluent activities in this study are shown to meet the shielding design limit given in Table 1.

V. Conclusions

A preliminary shielding design has been performed and evaluated. Tentative design criteria were established for design evaluation. The performance of the shielding design was compared against the shielding design criteria.

The results indicate that the support barrel, upper grid plate, and other reactor structures meet both the maximum neutron fluence and DPA limits established in the shielding design criteria with sufficient margin. Activities of the air effluent in the PSDRS were also evaluated and are shown to satisfy the effluent concentration limits in 10 CFR Part 20[6].

In this work, the fast neutron fluence and DPA were used as a neutron exposure limit in parallel. It is found that maximum DPAs show larger margins than the case when the neutron fluence limit is used as a design criterion. Therefore, the use of DPA as a shielding design criterion allows for a more flexibility in the LMR shielding design.

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—References—

(1) Kim, Young-In, *et al.* : "Conceptual Design of KALIMER Uranium Metallic Fueled Core", *KAERI/TR*-1279/99, KAERI (1999).
(2) Rhoades, W. A., Emmett, M. B. : "DOS: The Discrete Ordinates System", *ORNL/TM*-8362, ORNL (1982).
(3) Kim, J. D., Gil, C. S. : "KAFAX-F22: Development and Benchmark of Multi-group Library for Fast Reactor Using JEF-2.2", *KAERI/TR*-842/97, KAERI, (1997).
(4) MacFralane, R. E., "TRANSX 2: A Code for Interfacing MATXS Cross-Section Libraries to Nuclear Transport Codes," *LA*-12312-*MS*, LANL (1993).
(5) Kim, J. D., *et al.* : "Calculation and Comparative Analysis of Fe-56 DPA Cross Sections Processed from Evaluated Nuclear Data Libraries," *Proc. of the Korean Nuclear Society Autumn Meeting* (1998).
(6) US NRC : Code of Federal Regulations, Title 10, "Energy", Part 20, "Standard for Protection Against Radiation".