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RADIATION SHIELDING CALCULATION FOR THE MOX FUEL FABRICATION PLANT MELOX

Y. K. Lee*, J. C. Nimal*, J. P. Degrange**, M. Chiron***

* Laboratoire d'Études de Protection et Probabilité
Service d'Études des Réacteurs et de Mathématiques Appliquées
Commissariat à l'Energie Atomique, CEA - DRN/DMT/SERMA/LEPP
Centre d'Études de Saclay, 91191 Gif-sur-Yvette Cedex, France

** Centre d'Étude sur l'évaluation de la Protection dans le domaine Nucléaire
CEPN, 92263 Fontenay-aux-Roses, France

*** USSI Ingénierie, 78180 Saint Quentin en Yvelines, France

ABSTRACT

Radiation shielding calculation is an important engineering work in the design of the MOX fuel fabrication plant MELOX. Due to the recycle of plutonium and uranium from UO2 spent fuel reprocessing and the large capacity of production (120t HM/yr.), the shielding design requires more attention in this LWR fuel plant.

In MELOX, besides several temporary storage facilities of massive fissile material, about one thousand radioactive sources with different geometries, forms, densities, quantities and Pu concentrations, are distributed through different workshops from the PuO_2 powder reception unit to the fuel assembly packing room. These sources, with or without close shield, stay temporarily in different locations, containers and glove boxes.

In order to optimize the dimensions, the material and the cost of shield as well as to limit the calculation work in a reasonable engineer-hours, a calculation scheme for shielding design of MELOX is developed. This calculation scheme has been proved to be useful in consideration of the feedback from the evolutionary design and construction. The validated shielding calculations give a predictive but reliable radiation doses information.

I INTRODUCTION

Following the French policy of spent fuel reprocessing, plutonium recycling is of interest to reduce the natural uranium needs for pressurised water reactor (PWR) and to solve temporary the plutonium storage problem due to the delay in the fast breeder reactor (FBR) program.

The feasibility of recycling mixed oxide (MOX) fuel with a maximum ratio of 30% MOX assemblies in each reload has been demonstrated in French PWR plants since 1987. Now 16 Electricité de France (EDF) 900 MWe PWR have been authorised to accept the plutonium recycling and 5 EDF reactors use MOX fuel on a regular basis [1].

To satisfy the demand and to increase the MOX fuel fabrication capacity in France, a 120t HM/yr. plant MELOX has been constructed and is under commissioning at Cogema's Marcoule site.

With a high degree of automation design to operate the plant without the permanent presence of operators during the normal operation condition, the radiation dose to personnel may be largely reduced. Under normal operating conditions the design limit of the maximum annual individual radiation dose to personnel is set well below the international standards [2, 3].

In order to meet the requirements of radiation dose limits, the radiation shielding designs, calculations and radiation dose management require more attention for the quality control operations as well as for the maintenance and the minor incidental interventions [3].

Besides several temporary storage facilities of massive fissile material, about one thousand radioactive sources are found from the reception of PuO_2 to the delivery of fuel assembly. In this paper we present the principles of the calculation scheme developed for shielding design of MELOX in order to optimize the size, the thickness, the material and the cost of shield as well as to limit the calculation work in a reasonable engineer-hours.

II BACKGROUND

The MOX fuel fabrication process [3,4] is composed of the following steps: 1) reception and storage of raw materials; 2) blending of oxide powders; 3) preparation and sintering of fuel pellets; 4) grinding and checking of pellet dimensions; 5) loading pellets into cladding tube and fabrication of fuel pins; 6) fuel pins testing, fabrication of fuel assemblies and delivery of fuel assemblies.

In comparison with the traditional UO_2 fuel fabrication the raw materials and the blending processes in MELOX are completely different. Firstly, instead of using enriched UO_2, the PuO_2 and the tail uranium or eventually reprocessed uranium oxide powders are used in MELOX. Secondly, advanced blending processes [4] are needed for MOX fuel fabrication, namely with master blend to obtain micronized mixture powder and with secondary blend to obtain a specified ratio of Pu / (U+Pu).

Along the production line several temporary storage facilities are designed in the plant to store the raw materials, semi-products and products of the above fabrication processes.

In consideration of the isotopic composition of plutonium the following points related to source terms definition in this study are important for shielding design:

1) Plutonium is recycled from the UO_2 spent fuel at French reprocessing plant La Hague;
2) The burnup of UO_2 spent fuel before reprocessing is 33 GWd/t;
3) Plutonium is six years old after the reprocessing in order to limit the Am241 content;
4) The reference PuO_2 concentrations change from one hundred to a few percents corresponding to the content in raw material, master blend, secondary blend and fuel pins. Different Pu concentrations in fuel pins permit to zone [5] the MOX fuel assembly to minimize the impact on power peaking factor.

Besides plutonium aged three years and a half and recycled from UO_2 spent fuel with a 45 GWd/t burnup is also acceptable in MELOX.

Due to the large capacity of production (120t HM/yr.), the multiplication factor Keff for temporary storage facilities and for sources with high concentration and/or large quantity of Pu is carefully determined by French criticality group SEC/CEA by using Monte Carlo code MORET. The Keff value is useful in shielding calculation to prepare the energy spectrum and the intensity of neutron source.

From the practical radiation protection point of view, the following neutron and gamma sources from Pu and/or reprocessed U should be paid more attention:

1) neutron production from spontaneous fission and induced fission;
2) neutron production from (α,n) interaction (with a hard energy spectrum);
3) gamma ray from Am241 (60 keV), daughter of Pu241, for bare or less shielded source;
4) gamma ray from Tl208 (2.6 MeV), daughter of Pu236 and U232.

To facilitate the radiation shielding design and to assure the safe sub criticality condition, the general design criteria are given:

1) strengthened concrete walls separate the storage facilities from the production equipments;
2) borated fast neutrons shielding materials are used to separate storage units in the storage facilities to reduce the Keff value and neutron leakage;
3) inside the glove boxes, opaque close shields, fixed near or around the radioactive source, may be considered to reduce the radiation leakage from the most important sources;
4) at the outside surface of the glove boxes, transparent or opaque distant shields may protect the operators directly.

Normally the fire protection criteria should be considered for the close shield and distant shield, and for the neutron shielding materials between the storage units.

The location and the dimensions of distant shield depend not only on the different positions of operators but also on the expected duration and frequency of their interventions. The fixed or mobile shield should be effective for neutrons as well as for gamma rays.

To simplify the representation three different contributions and related evaluation methods for radiation dose are distinguished:

1) direct radiation from real sources, whose radiation dose is determined by point kernel integration calculation;
2) scattered radiation from real sources, whose radiation dose is evaluated by hall-scattering-albedo calculations;
3) radiation leakage from storage facilities, whose radiation dose is determined by advanced shielding codes.

Radiation leakage from neighbouring workshops is evaluated by consolidating the direct and scattered radiation contributions of the corresponding sources.

To comply the safe operation, in MELOX the design criteria concerning the dose equivalent rates are given by the following terms [3]:

1) in normal operation (ONF), for
   annual individual dose equivalent and
   annual averaged dose at 1 meter from glove boxes;
2) in incident condition (IIF), for
   averaged dose rate of intervention and
   maximum instantaneous dose rate.

It should be noted that only the external radiation dose has been treated in present study.

III CALCULATION SCHEME

To simplify the shielding design and radiation dose management, several workshops are firstly defined according to the functional units such as reception, processing, fabrication and delivery stations in the MELOX plant.

For each workshop, the radioactive sources are classified as following:

1) real source: it means any radioactive source in the workshop, for example a jar with 40 kg master blend at position A and an another jar with 20 kg master blend at position B. The two jars are identified as two real sources. A temporary storage facility is also considered as a real source.

2) special source: one kind of real source with complex structure which can not be treated directly by point kernel integration code. Most special sources are storage facilities.

3) standard source: one kind of real source defined by a quantity of fissile material, its container and eventually its close shield. It can be evaluated by the point kernel calculation. Standard sources are representative sources of the thousand real sources appearing in the plant. If a jar with 60kg master blend (at position C) are firstly defined as a standard source, the dose information of two real sources in positions A and B cited in above 1) can be directly evaluated from the dose rate data file of this standard source at C.

4) ACORA source: these real sources are defined by their geometrical and physical properties for calculations in ACORA-I programs. It can be a standard source or not. For example the three real sources cited above in positions A, B, and C are three ACORA sources.

After the identification of different source types in the beginning of project, four different calculation processes are used to achieve the objective of this study.

Firstly the shielding design of large storage facilities for fissile materials, namely special sources, are done to create dose rate data files containing the neutron and gamma dose information around the storage facilities.

Secondly the neutron and gamma dose rates data files are established for about one hundred representative radioactive sources, namely standard sources, selected from one thousand real sources.

Thirdly the neutron and gamma ray dose rate data files of standard source are exploited to construct the elementary dose rates of each ACORA source at identified calculation points.

Finally the design and calculation work for each workshop concerns the analysis of radiation doses related to different sources, shields and tasks according to the dose rate data base of above defined special sources and ACORA sources, the production plans and the dose limits.

The figure 1 shows a flow diagram of the calculation scheme developed for the radiation shielding design in the MELOX project.

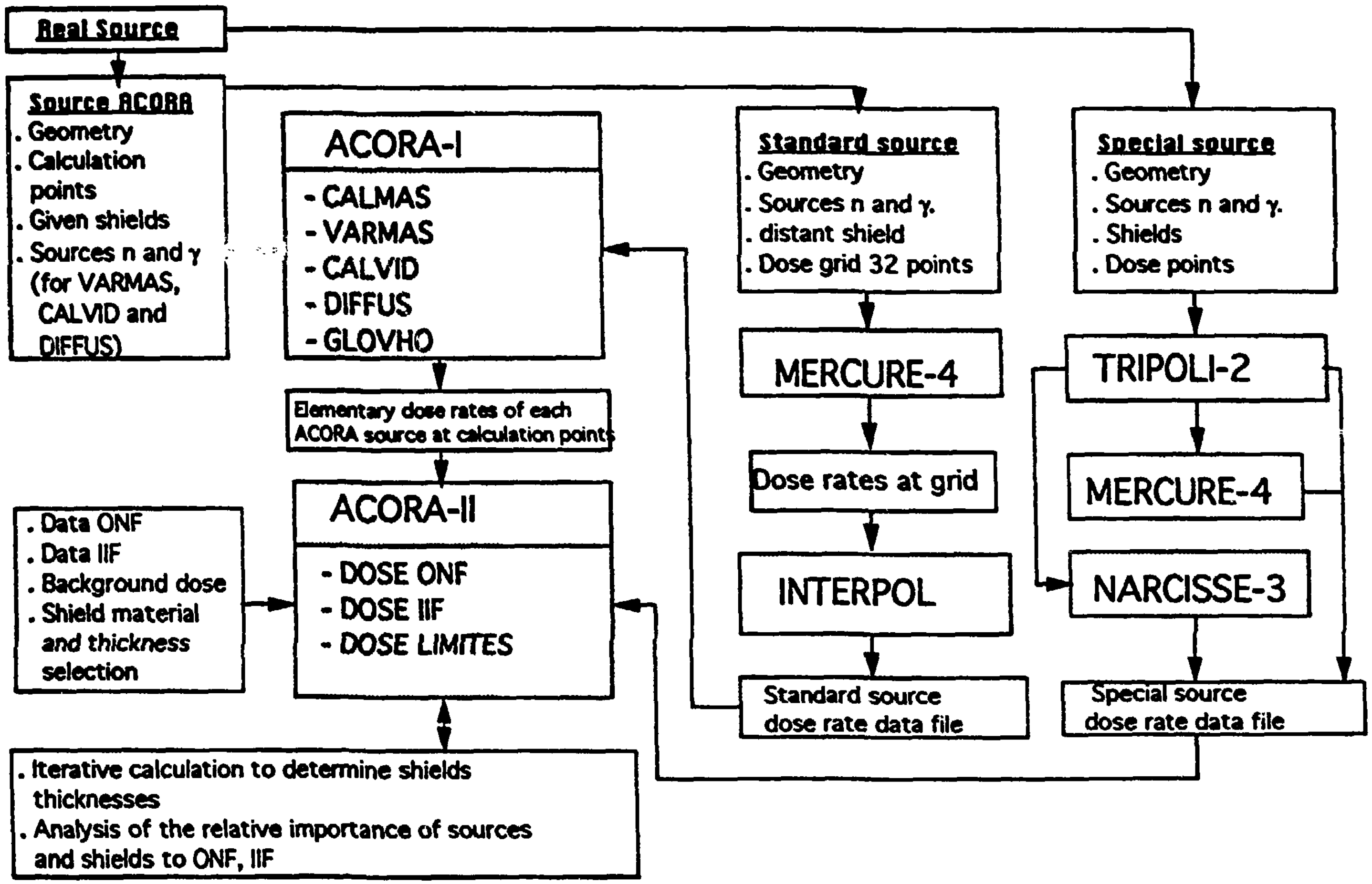
The traditional radiation shielding computer codes TRIPOLI-2 [6], MERCURE-4 [7] and NARCISSE-3 [8], are used alternately and iteratively to optimize the shielding designs and to evaluate the radiation doses around the storage facilities.

Several interpolation and extrapolation programs INTERPOL, CALMAS and VARMAS developed for this project allow us to construct the dose rate data base from limited calculations of MERCURE-4 and by using ACORA-I, ACORA-II programs it is possible to optimize conveniently the shields without reusing the MERCURE-4 shielding code.

A simple description of the main shielding codes used in this project is given in the following paragraphs. A description in detail of programs ACORA-I and ACORA-II is given in section V.

TRIPOLI-2 is a three dimensional Monte Carlo transport code with repeated geometry structures and with a very powerful and user friendly biasing scheme. For shielding application, a cross section library from ENDF/B4 with 315 groups for neutrons and 75 groups for gamma is used.

Fig.1 Calculation scheme for radiation shielding design of MELOX



MERCURE-4 is a three dimensional point kernel code which performs a stochastic integration by Monte Carlo method over the source volume with an economical CPU time. A cross section library with 17 groups for gamma ray is used. For neutron calculation, one group for fast neutron is considered.

To determine the dose response function and the remove cross sections of different distant shielding materials for neutrons used in MERCURE-4 calculations, the Sn code SN1D [9] and the MERCURE-4 code were iteratively used.

SN1D is a one dimensional Sn transport code, working in the modular environment. Its cross section preparation work is very user friendly. A 100 neutron and 30 gamma group VITAMIN C cross section library is available.

NARCISSE is a three dimensional hall-scattering code by using albedo concept. This code is used to evaluate the reflected neutron and gamma dose from roof and shielding walls.

IV SHIELDING CALCULATIONS FOR FISSILE MATERIAL STORAGE FACILITIES (SPECIAL SOURCES)

The storage facilities are normally designed for intermediate storage of raw materials, semi-products, products and wastes in the plant.

Under the full but sub-critical storage conditions, the shielding calculation and optimisation of the following facilities have been done:

1) after reception of raw materials, storage of AA227 containers, each containing five PuO_2 powder cans;
2) storage of PuO_2 powder in cans at production line before the blending process;
3) storage of master and secondary PuO_2-UO_2 powder mixture in jars at tunnel;
4) storage of fuel pellets in boxes before and after sintering;
5) storage of fuel pins horizontally on trays;
6) storage of fuel assemblages vertically at galleries;
7) storage of technical and operational waste containers on layered shelves.

The right-hand part of figure 1 shows the flow diagram of the calculations used to construct the dose rate data files of special sources.

TRIPOLI-2 code is used to calculate the neutron flux and gamma spectrum leaving the storage unit and sometimes to evaluate directly the neutron and gamma dose rate at pre-defined points behind the shield.

For each large storage facility of fissile material and waste installed in the plant, the complicated geometry of the entire storage facility is characterised by its repeated geometry structure in the central storage zone, and normally the total storage zone is surrounded by the outer strengthened shielding and civil structure wall.

It means that instead of treating the entire geometry of storage facility, the use of an unit storage geometry and reflection and/or translation options at unit storage boundaries in the TRIPOLI-2 simulation allow us to simplify the storage calculations.

For example a parallelepipedic unit storage cell with six reflection and/or translation boundary conditions in TRIPOLI-2 simulation represents the calculation of the infinite geometry of the corresponding three dimensional repeated storage unit.

To avoid the potential overestimate of the result, the reflection and/or translation boundary conditions in one or two sides of one to three directions among x, y and z axes are carefully chosen according to the real design of storage facilities.

The calculated neutron flux and gamma spectra leaving the storage unit defined in TRIPOLI-2 repeated geometry will be correct if the number of storage unit is large enough and otherwise conservative.

The advantage of using a repeated geometry in TRIPOLI-2 is not only to simplify the preparation work of input data but also to treat the neutron and gamma streaming from gaps or voids presented in the storage unit cell (For example the head part of fuel pins storage trays is plugged by a neutron shielding block to reduce the neutron leakage to nearby operational gallery. The gap between two trays is thus not negligible for radiation streaming.).

The TRIPOLI-2 code can also correctly treat the neutron shielding between the storage units and the radiation reflection from the floor and roof of the storage unit cell (For example the fuel assemblies are stored in a not fully shielded hall, and the reflection contribution is important to nearby packing room.).

For some storage facilities TRIPOLI-2 or NARCISSE codes are sequentially used, after the preliminary calculation of a one, two or three dimensional infinite storage unit by TRIPOLI-2, to evaluate the neutron and gamma dose rate at some points behind the shielding wall where the streaming and/or the scatter effects, due to the cavities, fire protection doors, labyrinths or ventilation openings ..etc., are important.

For most storage facilities, the doses and the complementary shields around the entire storage facility can be iteratively optimized and evaluated by the MERCURE -4 code by using the neutron flux and gamma spectra obtained from TRIPOLI-2 unit storage cell calculations as the source terms.

V REPRESENTATIVE SOURCES (STANDARD SOURCES) AND THEIR DOSE RATES DATA BASES

The central part of figure 1 shows a flow diagram of the calculations preparing the dose data files of standard sources.

The real radioactive sources are distributed through different workshops from the reception PuO_2 and UO_2 powder to the delivery of fuel assembly.

With different forms (powder, pellet, contaminated object ..etc.), densities (up to 10.4 g/cm^3 for fuel pellet), quantities (a few grams to several hundred kilograms) and Pu concentrations (30% PuO_2 for master blend and 10% for secondary blend), these sources, with or without close shield, stay temporary in different containers such as cans, pots, boxes, jars, funnels, claddings, transport casks and contaminated glove boxes.

To simplify the shielding designs and reduce the calculations works and cost, about one hundred representative radioactive sources were identified and selected as 'standard sources' among the thousand real sources anticipated in the fabrication lines and waste treatment lines.

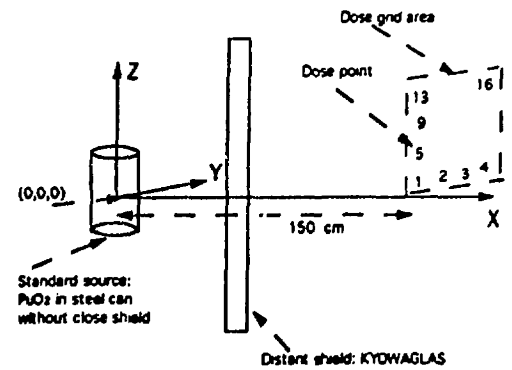
To obtain a flexible and rapid analysis of radiation doses and distant shields around the real radioactive sources and glove boxes, a data file of neutron and gamma radiation dose rate for each standard source is established by using the MERCURE-4 code and the interpolation code INTERPOL.

For each standard source several MERCURE-4 calculations - with two reference thicknesses for each distant shielding material, two to four directions around the standard source according to its symmetrical characteristics and four to six possible shielding materials - are done to calculate the neutron and gamma dose rates in a predetermined grid behind the distant shield.

For each direction and distant shielding material, the grid dose rates associated with the two reference shielding thicknesses from MERCURE-4 code are then used as the input data for the INTERPOL code which, developed for this project, allows us to construct the dose rate data files by interpolation and extrapolation for different thicknesses of the distant shield.

For example for a standard source with a cylindrical geometry, two different reference thicknesses of 2 and 6 cm are chosen in MERCURE-4 calculations for a transparent distant shield KYOWAGLAS placed in the radial direction then the INTERPOL calculation can produce a fine dose rate table for each 0.5 cm increment of thickness from 0 to 12 cm at every dose point of the dose grid defined in the MERCURE-4 calculations. If the radioactive source presents geometrical symmetry in axial and radial directions, the reduction of the dose grid area in MERCURE-4 calculations permits to reduce the CPU time. (Figure 2)

Figure 2. MERCURE-4 and INTERPOL calculation geometry



VI RADIATION DOSES ANALYSIS AND DISTANT SHIELD OPTIMISATION

The left-hand part of figure 1 shows the overall calculation scheme of ACORA. The ACORA code includes several developed programs, CALMAS, VARMAS, CALVID, DIFFUS and GLOVHO to calculate the elementary dose rates of each ACORA source and subsequently several simple summation programs, DOSEONF, DOSEIIF and DOSELIMIT to analyse the dose contributions and to optimize the distant shields.

A description of programs ACORA-I and ACORA-II is given in the following paragraphs:

ACORA-I

CALMAS: According to the location, the physical characteristics of the source and the planned distant shield, following calculations steps are used to obtain the instantaneous dose rate at a given work location from the corresponding standard sources dose rate data files:

1) coordinate transformations by rotation and translation from real geometrical situation in workshop to MERCURE-4 geometrical description;
2) localisation of the calculation point into the corresponding mesh of the dose grid of MERCURE-4 calculations;
3) the correction of the solid angle in consideration of the distance between the mass centre of source and the dose point;
4) the interpolation or extrapolation of the distant shield thicknesses;.
5) the interpolation or extrapolation of the dose grid.

This program is very useful to take into account the working plan changes, such as the shift of the positions of radioactive sources and/or operators in different design versions. It is not necessary, in such cases, to reuse the point kernel integration.

VARMAS: Some sources with varied but similar physical characteristics to one standard source, this subroutine may be used to calculate dose rates from the dose rate data file of that standard source. The variation of following parameters including mass, PuO_2 concentration, Keff, dimensions and density of the real source are considered.

This option is very effective to reduce the number of standard sources. For example a jar with 60 kg master blend is identified as a standard source then the dose rates around the same jar with less or more 60 kg master blend or secondary blend are easily evaluated by VARMAS.

In another example, the PuO_2 concentration is firstly fixed to 10% for all standard sources from secondary blend to fuel assembly, then the change of design comes with the 7.2% of PuO_2 concentration for secondary blend. The impact of this change and the related sensitivity analysis of dose rates and shields may be rapidly evaluated. The previously established dose rate data base (10% of PuO_2) of standard sources is still valid. Only the complementary calculations of VARMAS and ACORA-II are needed instead of recalculation of the MERCURE-4 and INTERPOL programs for all the related standard sources.

DIFFUS: The calculation of dose contribution from reflected neutron and gamma ray from the roof of workshop is necessary because:

1) For workshops with glove boxes, the roof of glove boxes are only specially shielded for 60keV photons from Am241.
2) For workshops without glove boxes, many sources with important mass are not fully or sufficiently covered.
3) For dose points distant from the sources, for example 5 to 10 meters, the direct radiation contribution of the dose is frequently not dominant.

This program is written by considering the following steps:

1) A reference source with fixed physical properties is selected.
2) The variation of neutron and gamma ray intensity for other sources is determined by considering the changes of mass, Pu concentration, Keff, density and dimensions.
3) The self shielding effect of the source is evaluated.
4) Some attenuation tables pre established by MERCURE-4 for neutron and gamma ray are used to calculate the upper shield.
4) The albedo concept is used to simplify the interactions inside the roof.
6) The collimating effect by a massive shield or nearby concrete wall is also included.

CALVID: This subroutine calculates analytically and directly the neutron and gamma doses rate by integrating an equivalent rectangle of considered source. The physical properties of reference source and the attenuation tables of shielding materials are identical to the DIFFUS subroutine.

This program is of interest to simplify the dose evaluation process as the attenuation table of the shielding materials is established. For some cases the uncertainty of calculated result is relative large but still acceptable in comparison with CALMAS and VARMAS calculations.

GLOVHO: Before the fabrication of fuel pins, equipments are housed in interconnected glove boxes. To permit maintenance, multitudes of gloves are provided at strategic points around the boxes and at regular intervals along the various transfer galleries.

Leaded curtains have been designed to attenuate the 60keV gamma ray. The neutron and hard gamma ray streaming through those glove holes is approximately evaluated for each dose point by weighting the dose rates, derived from previously established dose rate data files of related standard sources, by the fraction of solid angle intercepted by the glove holes.

ACORA-II

In ACORA-II phase [10], for each workshop, the radiation dose analysis and distant shield optimisation is iteratively done by determination of the annual individual dose equivalent and the average dose rate of intervention tasks according to:

1) the neutron and gamma doses contribution of each involved source and storage;
2) the presence of the different sources in each task;
3) the frequency, duration and location of each task;
4) the normal operation or incident conditions;
5) the limits related to dose rate, shield thickness and material ..etc..

The description in detail of DOSEONF, DOSEIIF and DOSELIMIT can be found at reference 3 and 10.

VII CONCLUSIONS

A radiation shielding calculation scheme for MOX fuel fabrication plant MELOX is presented. The traditional shielding codes like TRIPOLI-2, MERCURE-4, SN1D and NARCISSE are used to calculate the neutron and gamma dose and to optimize the shields around the storage facilities installed in the plant.

The point kernel code MERCURE-4 and the interpolation program INTERPOL are linked to establish the dose rate data base for about one hundred representative sources appearing in production lines and waste treatment unit in MELOX.

Using the established data base together with the interpolation programs like CALMAS, VARMAS, DIFFUS and the dose analysis program ACORA-II, the simplification of shielding calculation, the optimisation of the distant shield and the various analyses have been performed for engineering design of MELOX.

When the feedback from the evolution of designs and construction comes and when the constraints given by engineering and safety criteria related to fire protection, criticality and anti seismic change, the calculation scheme and its data base have been proved to be useful and effective.

The validation of the developed computer programs and calculation scheme has been done with Monte Carlo code TRIPOLI-2, and the calculation results have been proved to give a predictive but reliable radiation dose information.

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