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GENERATION OF LUMPED FISSION PRODUCT CROSS SECTIONS FOR HIGH BURNUP, HIGHLY ENRICHED URANIUM FUEL*

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A goal of the Advanced Neutron Source (ANS) Project is to design a reactor which is capable of generating a thermal neutron flux of at least 5×10^{19} n/m^2 sec in a reflector external to the reactor core. Several tentative designs which achieve this goal have been developed. An annular core composed of two fuel elements is being developed at Oak Ridge National Laboratory. [Ref. 1] To estimate the reactor's performance as a function of time, it is necessary to model fuel depletion and the buildup of fission product poisons. An exact calculation of fission product buildup is not feasible since there are several hundred fission product nuclides and concentrations would have to be computed at a few hundred points in the reactor. Consequently, a simplified treatment in which a few fission products are represented explicitly and the remainder grouped into two lumped fission products must be utilized.

The first set of reactor design calculations [Ref. 1] for the reactor design considered here were performed with a depletion methodology developed in 1966 [Ref. 2] for converter reactor studies. These analyses showed that the ANS reactor would have a cycle length of 14 days when operated at a power level of 270 Mw. Since both the cycle length and the discharge fuel burnup (209,000 MWD/MT) are very different from any of the reactors for which the depletion methodology was developed [Ref. 3], a new study of the depletion process was initiated.

Since the expected cycle length and fuel loading (18.1 kg ^{235}U) were known, input for an ORIGEN [Ref. 4] calculation could be prepared. For the work described here, cross section updates for the actinides and major fission products (those specified in Ref. 1) were prepared with data from an ENDF/B-V-derived library [Ref. 5]. The NITAWL-S and XSDRNPM-S codes were used to perform this update [Ref. 6,7]. The XSDRNPM model was a one-dimensional, buckled, cylindrical representation of the reactor described in Ref. 1. Fission yield values were derived from ENDF/B-IV data as contained in the ORIGEN Pressurized Water Reactor library.

The ORIGEN output was examined and 158 fission product nuclides were identified as being present in the core at a level greater than 10^{-5} moles. Of this set, cross-section data (27 group library derived from ENDF/B-V) were available for 127 nuclides.

To maintain consistency with prior work [Ref. 2], cross-section data for 17 nuclides which did not meet the aforementioned criteria were added to the fission product library. End-of-core-life concentrations for each of the 144 fission product nuclides were added to a one-dimensional, end-of-core-life (13.7 kg ^{235}U) reactor model. Using this model as input to the XSDRNPM code, absorption rates for all 144 nuclides were calculated. The nuclides were then grouped into three categories by establishing absorption rate limits which yielded groupings similar to those derived in Ref. 2. Those nuclides with absorption rates greater than 10^{14} absorptions/cm sec (absorption rate per unit height) would be modeled explicitly in the reactor calculations. Those with absorption rates from $1(10^{13})$ to $1(10^{14})$ absorptions/cm sec would be lumped into a moderately absorbing fission product and those with rates less than $1(10^{13})$ would be lumped into a weakly absorbing fission product.

The nuclide chains which constitute the explicitly modeled fission products are shown in Fig. 1. The differences between Fig. 1 and the explicitly modeled fission products in Ref. 2 include the deletion of the krypton production chain, the inclusion of the relatively short-lived

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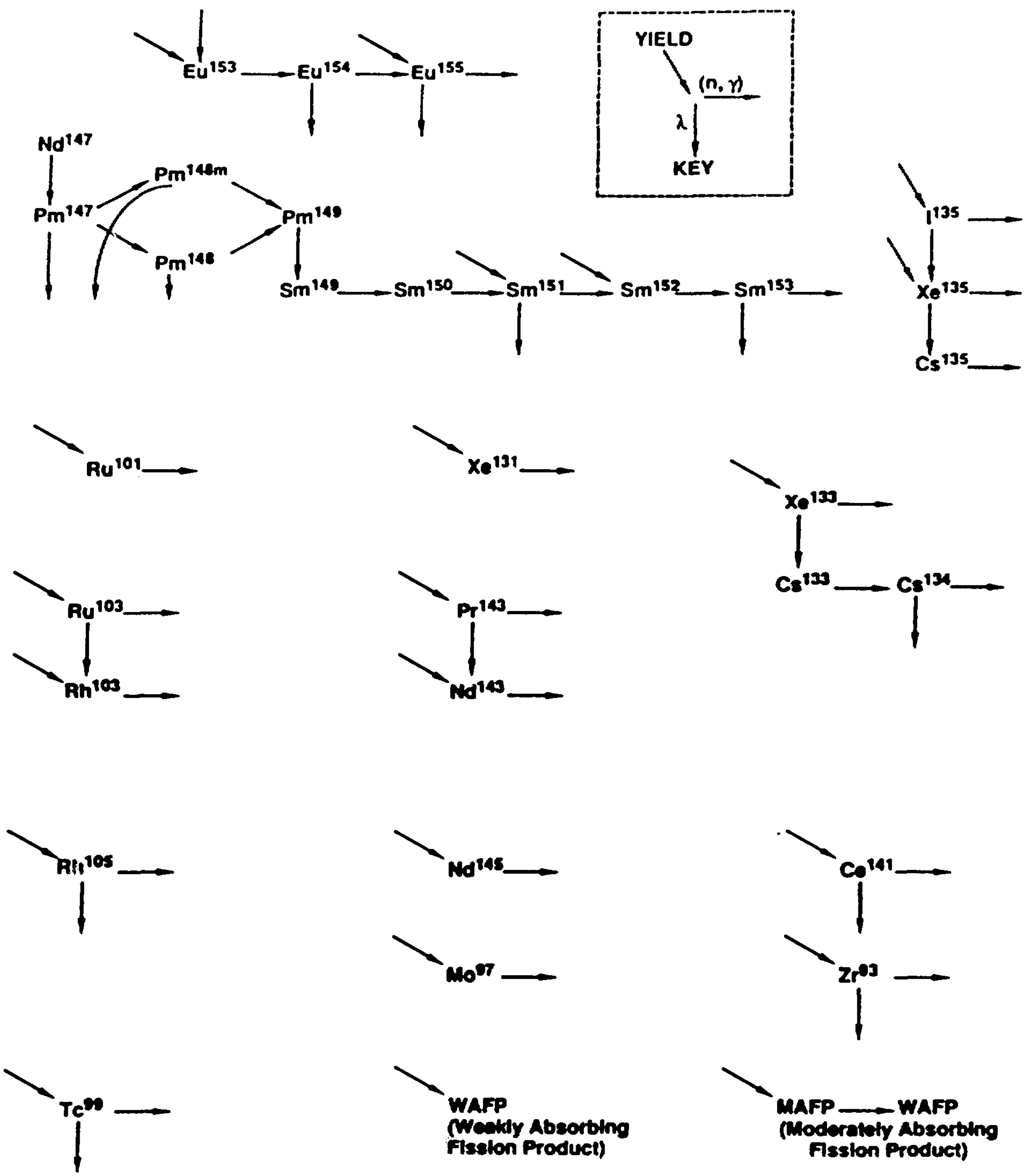


Fig. 1 Explicity Modeled Fission Product Chains

isotopes Ru-103, Pr-143, and Ce-141, and the inclusion of four longer-lived isotopes Ru-101, Mo-97, Tc-99 and Zr-93. The krypton production is significantly less important for ANS calculations than in previous work [Ref. 3] due to a reduction in Kr-82 fission yield from .0028 to .0001351. The short-lived isotopes are important in the current work because of the short, high-power density fuel cycle. While detailed information regarding the library used in the earlier work is not available, the inclusion of the four longer-lived isotopes as explicitly modeled nuclides probably reflects revised fission yields or cross sections.

To obtain lumped fission product cross sections which match the group structure of an ENDF/B-V derived library currently under development [Ref. 8], group structures of individual nuclides in an existing ENDF/B-V derived cross-section library were modified using a simple interpolation procedure. Nuclides from the resulting library were combined according to the fission yields. That is, if y_i is the yield for nuclide i, then the effective yield fraction for each lumped pseudo-element is

$$\bar{y} = \sum_i y_i \ ,$$

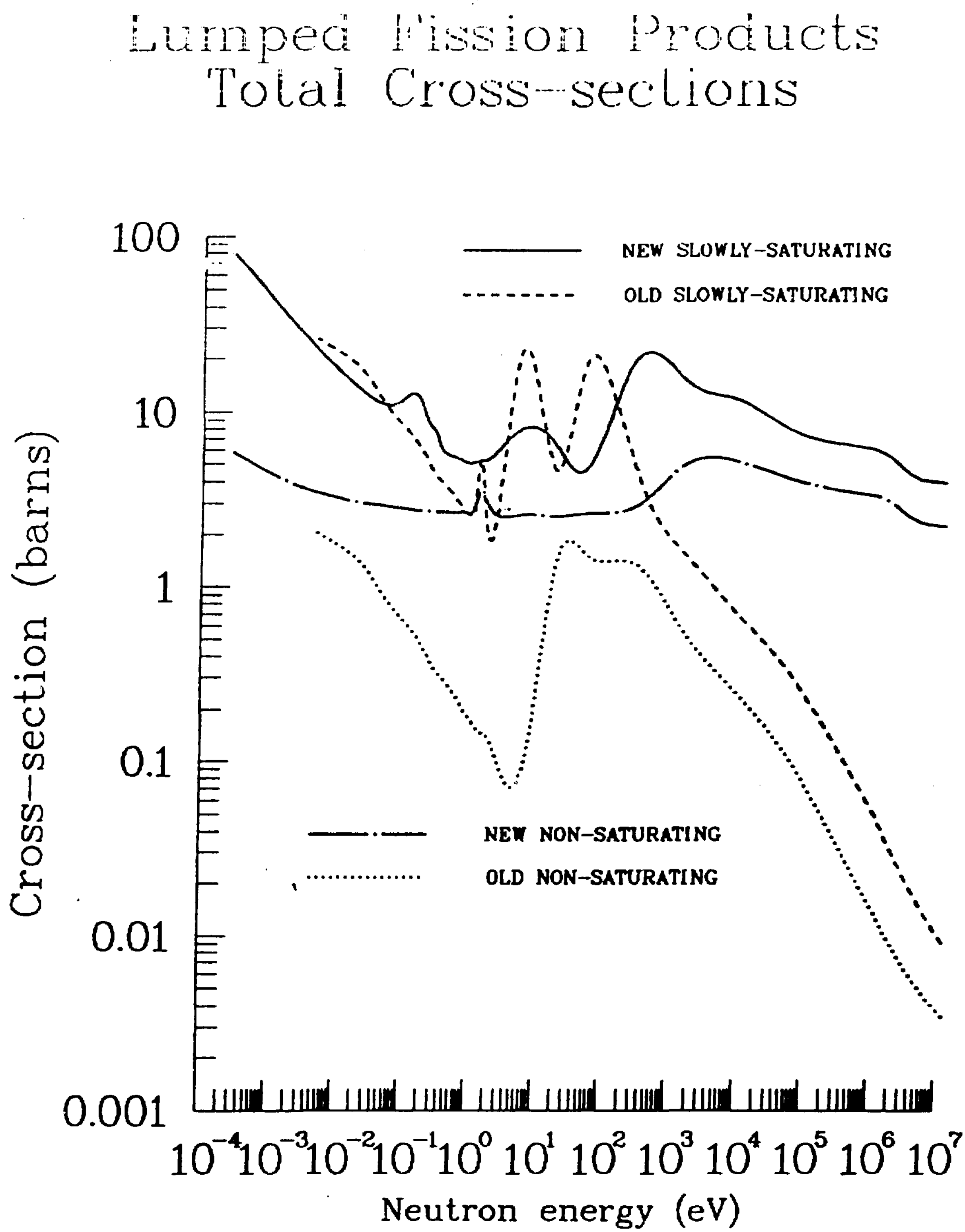
and the effective cross section for the lumped nuclide is

$$\sigma = \frac{1}{\bar{y}} \sum_i y_i \sigma_i \ .$$

The ICE code [Ref. 9] was used to perform this procedure. The total cross sections for the two lumped fission products generated by the code are shown in Fig. 2.

Lumped fission products for a short length, high burnup, highly enriched U-235 fuel cycle have been developed. Previous data were inade-

Figure 2



Lumped Fission Products
Total Cross-sections
NEW SLOWLY-SATURATING
OLD SLOWLY-SATURATING
NEW NON-SATURATING
OLD NON-SATURATING
100
10
1
0.1
0.01
0.001
Cross-section (barns)
10^-4 10^-3 10^-2 10^-1 10^0 10^1 10^2 10^3 10^4 10^5 10^6 10^7
Neutron energy (eV)

quate due to revised nuclear data, changes in expected reactor fuel cycle, and need for data in energy ranges previously not investigated.

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