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JAPANESE LIST OF REQUESTS FOR NUCLEAR DATA

April 1981

Compiled by

Sin-iti IGARASI
and
WRENDA Group of Japanese Nuclear Data Committee

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Japanese List of Requests for Nuclear Data

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Requests for nuclear data at 1980 are presented. They are 63 for fission reactor, 61 for fusion reactor, and 41 for safeguards. These will be registered in WRENDA 81/82+). This report contains these 165 requests, and also 111 requests which were withdrawn from WRENDA. These withdrawn requests are 39 for fission reactor, 44 for fusion reactor and 28 for safeguards.

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Keywords : Nuclear Data, Data Requests, Fission Reactors, Fusion Reactors, Safeguards, WRENDA

+) WRENDA: World Request List for Nuclear Data, see Ref. 1.

核データに対する日本の要求リスト

シグマ研究委員会 WRENDA グループ *)
五十嵐信一（編）

（1981年4月3日受理）

1980年の時点での核データの要求リストをまとめた。ここには核分裂炉関係63件，核融合炉関係61件，保障措置関係41件の要求を載せた。これらの要求は，WRENDA 81／82+) に登録されることになる。この報告書には，これら165件の要求リストの他にWRENDAから取り下げた111件のリストも載せた。この取り下げ分は核分裂炉で39件，核融合炉で44件，保障措置で28件になっている。

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+) WRENDA：World Request List for Nuclear Data のことである。参考文献1を参照。

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1. Introduction

While the number of the Japanese requests submitted to WRENDA 79/80[1] was 244, the Japanese requests of 245 are registered in the WRENDA 79/80. This is because the WRENDA 79/80 includes one additional request of the resonance parameters which was separated from the original request[2] of the ^{131}Xe capture cross section. So, the number of the previous Japanese requests are regarded as 245 in order to avoid any unnecessary confusion.

WRENDA Group of Japanese Nuclear Data Committee was convened in October 1980 to screen new requests, to examine the old Japanese requests and to compile Japanese List of Requests for Nuclear Data in 1980. About 60 new requests were received by the WRENDA Group, and were examined whether or not they were appropriate to the requests from the viewpoint of the present status of the nuclear data. When questions arose about the requests, the WRENDA Group asked the requestors the reasons behind their requests and discussed with them about the questionable matters of their requests. In some cases, the WRENDA Group modified partly the descriptions of the original requests so that they might fit the status quo of the data. Finally, the WRENDA Group adopted 20 requests for fission reactors and 11 for fusion reactors.

The old requests[1,2] were returned to the requestors and were reviewed in the light of the data status and needs. The data of the angular distributions for the elastic and inelastic scattering cross sections requested from fusion branch were mostly withrawn, because the energy-angle double differential cross sections are preferable for the fusion research to the so-called single differential cross sections. The new requests for fusion reactors mentioned above were mostly the double differential cross sections which took the place of the withdrawn differential data.

Many requests for the evaluated data were also withdrawn. This is due to penetration of the JENDL activity into the data users in Japan. Especially, there were many withdrawn requests in the mass region of the fission products. The new requests for the FP nuclides are increasing their accuracies and qualities.

In the present review work, 111 requests were finally withdrawn. Hence, Japanese requests of 165 will be registered in WRENDA. In this report, these requests are presented in the form of the computer output lists. JAERI/NDC made a new computer system[3)] of storage and retrieval for the nuclear data requests. This system will be used as an archives of Japanese requests for the nuclear data. Various kinds of retrievals are available from this system. Details on this new system will be presented elsewhere.

2. List of Requests Submitted to New WRENDA

In this section, Japanese requests submitted to new WRENDA are presented in the form of output lists from the computer. First line of each request shows, from the left, sequential number for this request list, atomic number, name of element, mass number, incident particle, and physical quantity, respectively. Second line gives generation figure, date of reception at JAERI/NDC, date of revision (if any, with increase of generation figure), date of submission to WRENDA, and date of withdrawal from WRENDA (for only withdrawn requests). Third line is for the energy region, required accuracy, priority, and requestors' names. If requestors are more than three persons, their names are shown in fourth and fifth lines. The next line is for the registration numbers for this storage and retrieval system and for WRENDA entry, and category for application. Comment area is devoted to quantity comment (Q), accuracy comment (A) and other comment (O), respectively. These structures are made so as to reproduce the WRENDA lists.

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                    ** REQUESTS REGISTERED TO WRENDA.        **                          PAGE    1

1 001 HYDROGEN 002      NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
  00  801110              801225
            UP TO    15.00 MEV    15.0%   2    OSA   A.TAKAHASHI
      800001           FUSION REACTOR
                 Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTION FOR (N,2N) REACTION WANTED.
                 O: FOR ESTIMATION OF EMITTED NEUTRON SPECTRA FROM D-T MIXTURE OF INERTIALLY
                    CONFINED TARGET PLASMA.

2 001 HYDROGEN 003      NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
  00  801110              801225
            UP TO    15.00 MEV    15.0%   2    OSA   A.TAKAHASHI
      800002           FUSION REACTOR
                 Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTION FOR (N,2N) REACTION WANTED.
                 O: FOR ESTIMATION OF EMITTED NEUTRON SPECTRA FROM D-T MIXTURE OF INERTIALLY
                    CONFINED TARGET PLASMA.

3 003 LITHIUM 006       NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
  00  760401              761010
          1.00 MEV    15.00 MEV    15.0%   2    JAE   Y.SEKI              MAP   M.KASAI
      780003  762054   FUSION REACTOR
                 O: GAMMA-RAY HEATING CALCULATIONS.

4 003 LITHIUM 006       NEUTRON                N,ND
  00  760401              761010
            UP TO    15.00 MEV    10.0%   2    JAE   Y.SEKI
      780004  762052   FUSION REACTOR
                 O: NEUTRONICS CALCULATIONS AND ENERGY DEPOSITION.

5 003 LITHIUM 006       NEUTRON                N,NT
  00  760401              761010
          3.00 MEV    15.00 MEV     5.0%   1    JAE   Y.SEKI
      780005  762053   FUSION REACTOR
                 O: TRITIUM BREEDING AND ENERGY DEPOSITION CALCULATION.

6 003 LITHIUM 006       NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
  00  801110              801225
          2.00 MEV    15.00 MEV    10.0%   2    OSA   A.TAKAHASHI         JAE   Y.SEKI
      800003           FUSION REACTOR
                 Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTIONS REQUIRED WITH INCIDENT ENERGY STEP OF
                    0.5 MEV.
                 O: NEUTRON TRANSPORT AND TRITIUM PRODUCTION RATE CALCULATIONS.  ANGULAR
                    DISTRIBUTIONS OF INELASTICALLY SCATTERED NEUTRONS FOR ALL AVAILABLE DISCRETE
                    LEVELS ALSO REQUIRED.

7 003 LITHIUM 007       NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
  00  760401              761010
        0.25-1 EV     15.00 MEV    15.0%   2    JAE   Y.SEKI
      780010  762059   FUSION REACTOR
                 Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                 O: GAMMA-RAY HEATING CALCULATIONS.
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 8 003 LITHIUM 007       NEUTRON                N,NT
   01  760401    801110    761010
            UP TO    15.00 MEV     5.0%   1    JAE   Y.SEKI              OSA   A.TAKAHASHI
       780013  762058   FUSION REACTOR
                   Q: NEUTRON SPECTRA WITH ACCURACY 15 PER CENT ALSO REQUIRED.
                   O: TRITIUM BREEDING AND ENERGY DEPOSITION CALCULATION.

 9 003 LITHIUM 007       NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
   00  801110              801225
           2.00 MEV    15.00 MEV    10.0%   2    JAE   Y.SEKI              OSA   A.TAKAHASHI
       800004           FUSION REACTOR
                   Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTIONS FOR TOTAL NEUTRON EMISSIONS REQUIRED.
                   O: NEUTRON TRANSPORT AND TRITIUM PRODUCTION RATE CALCULATIONS.  ANGULAR
                      DISTRIBUTIONS OF INELASTICALLY SCATTERED NEUTRONS FOR ALL AVAILABLE DISCRETE
                      LEVELS ALSO REQUIRED.

10 004 BERYLLIUM 009     NEUTRON                N,ALPHA
   00  760401              761010
           8.00 MEV    15.00 MEV    15.0%   3    JAE   Y.SEKI
       780018  762063   FUSION REACTOR
                   O: HELIUM ACCUMULATION CALCULATIONS.

11 004 BERYLLIUM 009     NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
   00  801110              801225
           1.70 MEV    15.00 MEV    15.0%   2    JAE   Y.SEKI              OSA   A.TAKAHASHI
       800005           FUSION REACTOR
                   Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTIONS FOR TOTAL NEUTRON EMISSIONS REQUIRED.
                      DOUBLE DIFFERENTIAL FOR THE (N,2N) REACTION IS ALSO REQUIRED BY A.TAKAHASHI.
                   O: BLANKET NEUTRONICS CALCULATIONS.  FOR ALSO NEUTRON MULTIPLICATION CALCULATIONS.

12 006 CARBON 012        NEUTRON                N,N 3ALPHA
   00  760401              761010
            UP TO    15.00 MEV    15.0%   2    JAE   Y.SEKI
       780020  762065   FUSION REACTOR
                   Q: TOTAL ALPHA PRODUCTION CROSS SECTION AND SECONDARY NEUTRON ENERGY SPECTRUM
                      REQUIRED.
                   O: NEUTRON TRANSPORT AND HELIUM ACCUMULATION CALCULATIONS.

13 006 CARBON 012        NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
   00  801110              801225
           7.00 MEV    15.00 MEV    10.0%   2    JAE   Y.SEKI              OSA   A.TAKAHASHI
       800006           FUSION REACTOR
                   Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTIONS FOR TOTAL NEUTRON EMISSIONS REQUIRED.
                      ANGULAR DISTRIBUTIONS OF INELASTIC SCATTERING CROSS SECTIONS FOR ALL AVAILABLE
                      DISCRETE LEVELS ESPECIALLY WANTED BY A.TAKAHASHI.
                   O: NEUTRON TRANSPORT CALCULATIONS.
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14 006 CARBON 012        NEUTRON                N,N  ALPHA
   00  801110              801225
             UP TO    40.00 MEV    15.0%   2    KTO   K.SHIN              TOH   H.ORIHARA
                                                NAG   S.ITOH
       800011            FUSION REACTOR
                   Q: SECONDARY NEUTRON AND ALPHA-PARTICLE ENERGY SPECTRA ARE REQUIRED.
                   O: FOR DETECTOR EFFICIENCY DETERMINATION IN FUSION REACTOR NEUTRONICS EXPERIMENTS.

15 006 CARBON 012        NEUTRON                N,P
   00  801110              801225
         5.00 MEV    20.00 MEV     5.0%   2    NAG   S.ITOH
       800031            FUSION REACTOR
                   O: FOR CALCULATION OF DETECTOR RESPONSE FUNCTION. DISAGREEMENT BETWEEN KREGER AND
                      RIMMER ABOVE 16.0 MEV.

16 006 CARBON 013        ALPHA                  NEUTRON EMISSION CROSS SECTION
   00  780808              781212
             UP TO    10.00 MEV    20.0%   2    SAE   N.YAMANO
       780021  792070   FISSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED.  ANGULAR DISTRIBUTION ALSO REQUIRED.  REQUIRED NEUTRON
                      ENERGIES ARE 100 KEV TO 10 MEV.
                   O: FOR NEUTRON SHIELDING AND EVALUATION OF NEUTRON SOURCE. FOR EVALUATION OF
                      NEUTRON ENERGY SPECTRUM IN FUEL RECYCLE PROCESS.

17 008 OXYGEN 016        NEUTRON                N,ALPHA
   00  760401              761010
         7.50 MEV    15.00 MEV    15.0%   2    JAE   Y.SEKI
       780022  762066   FUSION REACTOR
                   Q: TOTAL ALPHA PRODUCTION CROSS SECTION.
                   O: HELIUM ACCUMULATION CALCULATION IN LI-OXIDE BLANKETS.

18 008 OXYGEN 016        NEUTRON                N,N  ALPHA
   00  760401              761010
             UP TO    15.00 MEV    15.0%   2    JAE   Y.SEKI
       780023  762067   FUSION REACTOR
                   Q: SECONDARY NEUTRON ENERGY SPECTRA REQUIRED.
                   O: CALCULATION OF NEUTRON TRANSPORT AND HELIUM ACCUMULATION IN LI-OXIDE BLANKETS.

19 008 OXYGEN 016        TRITON                 NEUTRON EMISSION CROSS SECTION
   00  780808              781212
             UP TO    12.00 MEV    10.0%   2    JAE   K.TANAKA            JAE   H.KUDO
       780024  792071   FUSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED.
                   A: 5% ENERGY RESOLUTION DESIRABLE.
                   O: FOR PRECISE ESTIMATION OF LI2O BURNUP IN CTR BLANKET.  FOR EVALUATION OF NUMBER
                      OF O18 ATOMS FROM BETA PLUS DECAY OF F18 PRODUCED THROUGH O16(T,N)F18.
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20 008 OXYGEN 017        NEUTRON                N,ALPHA
   00  780808              781212
         0.25-1 EV     15.00 MEV    30.0%   2    MAP   T.KAWAKITA
       780025  792073   FISSION REACTOR
                   Q: EVALUATED DATA WANTED.
                   O: FOR EVALUATION OF QUANTITY OF C14 FROM OXIDE FUEL IN FAST REACTOR.  BOTH
                      EVALUATIONS AND MEASUREMENTS ARE SCARCE.

21 008 OXYGEN 017        ALPHA                  NEUTRON EMISSION CROSS SECTION
   00  780808              781212
             UP TO     10.00 MEV    20.0%   2    SAE   N.YAMANO
       780026  792072   FISSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED.  ANGULAR DISTRIBUTION ALSO REQUIRED.  REQUIRED NEUTRON
                      ENERGIES ARE 100 KEV TO 10 MEV.
                   O: FOR NEUTRON SHIELDING AND EVALUATION OF NEUTRON SOURCE.  FOR EVALUATION OF
                      NEUTRON ENERGY SPECTRUM IN FUEL RECYCLE PROCE.

22 008 OXYGEN 018        ALPHA                  NEUTRON EMISSION CROSS SECTION
   00  780808              781212
             UP TO     10.00 MEV    20.0%   2    SAE   N.YAMANO
       780027  792074   FISSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED.  ANGULAR DISTRIBUTION ALSO REQUIRED.  REQUIRED NEUTRON
                      ENERGIES ARE 100 KEV TO 10 MEV.
                   O: FOR NEUTRON SHIELDING AND EVALUATION OF NEUTRON SOURCE.  FOR EVALUATION OF
                      NEUTRON ENERGY SPECTRUM IN FUEL RECYCLE PROCESS.

23 008 OXYGEN 018        ALPHA                  TOTAL NEUTRON YIELD
   01  760401    801110    761010
         5.10 MEV      5.50 MEV     5.0%   2    PNC   S.SUZUKI
       780028  762041   SAFEGUARDS
                   Q: ABSOLUTE NEUTRON YIELD REQUIRED.
                   O: DETECTION OF PU BY NEUTRON COINCIDENCE METHOD.

24 009 FLUORINE 019      NEUTRON                INELASTIC CROSS SECTION
   00  760401              761010
         1.00 MEV     15.00 MEV    10.0%   3    JAE   Y.SEKI
       780029  762068   FUSION REACTOR
                   O: POTENTIAL CONSTITUENT IN COOLANT,FLIBE.  TRITIUM BREEDING CALCULATIONS.

25 009 FLUORINE 019      NEUTRON                ABSORPTION CROSS SECTION
   00  760401              761010
         0.25-1 EV     15.00 MEV    10.0%   3    JAE   Y.SEKI
       780030  762069   FUSION REACTOR
                   O: POTENTIAL CONSTITUENT IN COOLANT,FLIBE.  TRITIUM BREEDING CALCULATIONS.

26 013 ALUMINIUM 027     NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  760401              761010
         0.25-1 EV     15.00 MEV    15.0%   3    MAP   M.KASAI
       780032  762075   FUSION REACTOR
                   O: GAMMA-RAY HEATING CALCULATIONS.
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27 013 ALUMINIUM 027     NEUTRON                N,D
   00  760401              761010
             UP TO    15.00 MEV    15.0%   3    MAP   M.KASAI
       780035  762072   FUSION REACTOR
                   O: HYDROGEN ACCUMULATION CALCULATIONS.

28 013 ALUMINIUM 027     NEUTRON                N,T
   00  760401              761010
             UP TO    15.00 MEV    15.0%   3    MAP   M.KASAI
       780036  762073   FUSION REACTOR
                   O: HYDROGEN ACCUMULATION CALCULATIONS.

29 014 SILICON 030       NEUTRON                CAPTURE CROSS SECTION
   00  780808              781212
         1.00-4 EV    100.00 KEV    10.0%   3    JAE   N.AOYAGI
       780037  792075   FISSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED.
                   O: FOR DOPING P31 INTO SINGLECRYSTAL OF SI BY NEUTRON IRRADIATION TO MAKE
                      SEMICONDUCTOR. ONLY A FEW OLD DATA ARE AVAILABLE.

30 018 ARGON 040         NEUTRON                CAPTURE CROSS SECTION
   00  710401              711010
             UP TO    10.00 MEV            2    NIG   M.KAWAI
       780039  712006   FISSION REACTOR
                   A: ACCURACY REQUIRED TO BETTER THAN 20.0 PERCENT.
                   O: FOR REACTOR HAZARD CALCULATION.

31 019 POTASSIUM 039     NEUTRON                N,P
   00  780808              781212
          0.25-1 EV     15.00 MEV    30.0%   2    MAP   T.KAWAKITA
       780040  792076   FISSION REACTOR
                   Q: EVALUATED DATA WANTED.
                   O: FOR REACTOR HAZARD CALCULATION.  THERE ARE MANY EXPERIMENTAL DATA IN MEV REGION.

32 020 CALCIUM           NEUTRON                ELASTIC CROSS SECTION
   00  760401              761010
             1.00 MEV    15.00 MEV    15.0%   3    JAE   Y.SEKI
       780041  762234   FUSION REACTOR
                   O: INCLUDED IN CONCRETE.  SHIELDING DESIGN.

33 020 CALCIUM           NEUTRON                DIFFERENTIAL ELASTIC CROSS SECTION
   00  760401              761010
             1.00 MEV    15.00 MEV    15.0%   3    JAE   Y.SEKI
       780042  762076   FUSION REACTOR
                   O: INCLUDED IN CONCRETE.  SHIELDING DESIGN.
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34 020 CALCIUM            NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  760401               761010
         500.00 KEV     15.00 MEV     15.0%   3    JAE    Y.SEKI
       780044  762078    FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: INCLUDED IN CONCRETE.  GAMMA-RAY HEATING CALCULATIONS.

35 022 TITANIUM           NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  760401               761010
         0.25-1 EV      15.00 MEV     15.0%   3    MAP    M.KASAI
       780046  762083    FUSION REACTOR
                    O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  GAMMA-RAY HEATING CALCULATIONS.

36 022 TITANIUM           NEUTRON                N,ALPHA
   00  760401               761010
             UP TO      15.00 MEV     15.0%   3    MAP    M.KASAI
       780049  762082    FUSION REACTOR
                    O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  HELIUM ACCUMULATION CALCULATIONS.

37 023 VANADIUM           NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  760401               761010
         0.25-1 EV      15.00 MEV     10.0%   2    MAP    M.KASAI
       780052  762089    FUSION REACTOR
                    O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  GAMMA-RAY HEATING CALCULATIONS.

38 023 VANADIUM           NEUTRON                N,2N
   00  760401               761010
             UP TO      15.00 MEV     10.0%   2    MAP    M.KASAI
       780053  762085    FUSION REACTOR
                    O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  NEUTRON MULTIPLICATION
                       CALCULATIONS.

39 024 CHROMIUM           NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   01  760401    801110     761010
             UP TO      15.00 MEV     15.0%   2    JAE    Y.SEKI
       780059  762094    FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: GAMMA-RAY HEATING CALCULATIONS.

40 024 CHROMIUM           NEUTRON                N,2N
   00  760401               761010
             UP TO      15.00 MEV     15.0%   2    JAE    Y.SEKI
       780060  762095    FUSION REACTOR
                    O: NEUTRON BALANCE CALCULATIONS.
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41 024 CHROMIUM          NEUTRON                N,P
   00  760401              761010
            UP TO    15.00 MEV    20.0%   2    JAE   Y.SEKI
       780061  762096   FUSION REACTOR
                 O: HYDROGEN ACCUMULATION CALCULATIONS.

42 024 CHROMIUM          NEUTRON                N,ALPHA
   00  760401              761010
            UP TO    15.00 MEV    20.0%   2    JAE   Y.SEKI
       780062  762097   FUSION REACTOR
                 O: HELIUM ACCUMULATION CALCULATIONS.

43 026 IRON              NEUTRON                INELASTIC CROSS SECTION
   01  760401    801110    761010
            UP TO    20.00 MEV     5.0%   1    JAE   Y.SEKI              NIG   M.KAWAI
       780064  762099   FUSION REACTOR
                 O: INELASTIC GAMMA RAY SPECTRA ALSO REQUIRED.
                 O: NEUTRON TRANSPORT AND GAMMA-RAY HEATING CALCULATIONS.

44 026 IRON              NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  760401              761010
         0.25-1 EV     15.00 MEV    10.0%   2    MAP   M.KASAI
       780066  762104   FUSION REACTOR
                 O: GAMMA-RAY HEATING CALCULATIONS.

45 026 IRON              NEUTRON                N,2N
   00  760401              761010
            UP TO    15.00 MEV    10.0%   2    JAE   Y.SEKI
       780067  762101   FUSION REACTOR
                 O: NEUTRON MULTIPLICATION CALCULATIONS.

46 026 IRON              NEUTRON                N,P
   00  760401              761010
            UP TO    15.00 MEV    20.0%   2    JAE   Y.SEKI
       780068  762102   FUSION REACTOR
                 O: HYDROGEN ACCUMULATION CALCULATIONS.

47 026 IRON              NEUTRON                N,ALPHA
   00  760401              761010
            UP TO    15.00 MEV    20.0%   2    JAE   Y.SEKI
       780069  762103   FUSION REACTOR
                 O: HELIUM ACCUMULATION CALCULATIONS.
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48 026 IRON 000          NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
   00  801110              801225
            UP TO    15.00 MEV    10.0%   2    CSA   A.TAKAHASHI
       800007           FUSION REACTOR
                 Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTIONS FOR INELASTIC SCATTERING AND (N,2N)
                    REACTION ARE ESPECIALL WANTED.
                 O: NEUTRON TRANSPORT CALCULATION.

49 026 IRON 057          NEUTRON                INELASTIC CROSS SECTION
   00  801110              801225
            UP TO   800.00 KEV    10.0%   2    NIG   M.KAWAI
       800014           FISSION REACTOR
                 O: FOR REACTOR SHIELDING CALCULATION.

50 028 NICKEL            NEUTRON                INELASTIC CROSS SECTION
   01  760401    801110    761010
            UP TO    20.00 MEV     5.0%   1    JAI   Y.SEKI              MAP   M.KASAI
                                                NIG   M.KAWAI
       780072  762105   FUSION REACTOR
                 Q: INELASTIC GAMMA-RAY SPECTRA ALSO REQUIRED.
                 O: NEUTRON TRANSPORT AND GAMMA-RAY HEATING CALCULATIONS.

51 028 NICKEL            NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  760401              761010
         0.25-1 EV      15.00 MEV    10.0%   2    MAP   M.KASAI
       780074  762111   FUSION REACTOR
                 O: GAMMA-RAY HEATING CALCULATIONS.

52 028 NICKEL            NEUTRON                N,2N
   00  760401              761010
            UP TO    15.00 MEV    15.0%   2    JAE   Y.SEKI              MAP   M.KASAI
       780075  762106   FUSION REACTOR
                 O: NEUTRON BALANCE CALCULATIONS.

53 028 NICKEL            NEUTRON                N,P
   00  760401              761010
            UP TO    15.00 MEV    20.0%   2    JAE   Y.SEKI              MAP   M.KASAI
       780076  762107   FUSION REACTOR
                 O: HYDROGEN ACCUMULATION CALCULATIONS.

54 028 NICKEL            NEUTRON                N,ALPHA
   00  760401              761010
            UP TO    15.00 MEV    20.0%   2    JAE   Y.SEKI              MAP   M.KASAI
       780078  762108   FUSION REACTOR
                 O: HELIUM ACCUMULATION CALCULATIONS.
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55 029 COPPER            NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  760401              761010
         0.30-1 EV     15.00 EV     15.0%   2    JAE   Y.SEKI
       780081  762113   FUSION REACTOR
                   Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                   O: GAMMA-RAY HEATING IN MAGNETS.

56 030 ZINC 064          NEUTRON                CAPTURE CROSS SECTION
   00  780808              780808
         0.25-1 EV     15.00 MEV    20.0%   2    MAP   T.KAWAKITA
       780082  792077   FISSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED.
                   O: FOR ESTIMATION OF RADIOACTIVITY OF SPENT STRUCTURAL MATERIALS IN FAST REACTORS.
                      BOTH EXPERIMENTAL AND EVALUATED DATA ARE SCARCE.

57 035 BROMINE 087                               GAMMA RAY YIELD
   00  760401              761010
                                      10.0%   3    TOS   H.SHIMOJIMA
       780083  762001   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 1419 KEV GAMMA-RAY REQUIRED. (FOLLOWING BETA DECAY
                      EVENT).
                   O: DETECTION OF FAILED FUEL.

58 035 BROMINE 088                               GAMMA RAY YIELD
   00  760401              761010
                                      10.0%   3    TOS   H.SHIMOJIMA
       780084  762002   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 767 KEV GAMMA-RAY REQUIRED. (FOLLOWING BETA DECAY
                      EVENT).
                   O: DETECTION OF FAILED FUEL.

59 036 KRYPTON 090                               GAMMA RAY YIELD
   00  760401              761010
                                      10.0%   3    TOS   H.SHIMOJIMA
       780085  762003   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF MAJOR GAMMA-RAYS REQUIRED. (FOLLOWING BETA DECAY
                      EVENT).
                   O: DETECTION OF FAILED FUEL.

60 040 ZIRCONIUM 093     NEUTRON                CAPTURE CROSS SECTION
   01  750401    801110    751010
         100.00 EV    500.00 KEV    20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780086  752004   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.
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61 040 ZIRCONIUM 093     NEUTRON                RESONANCE PARAMETERS
   01  780808    801110    781212
                                    20.0%   2    SAE   H.MATSUNOBU          NIG   S.IIJIMA
       780087  792068   FISSION REACTOR
                   Q: FOR FAST REACTOR BURNUP CALCULATIONS.  MORE RESONANCE DATA ARE REQUIRED.  ONLY
                      ONE RESONANCE LEVEL AT 110 EV. NO KEV DATA.

62 041 NIOBIUM 000       NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
   00  801110              801225
            UP TO    15.00 MEV    10.0%   2    OSA   A.TAKAHASHI
       800009           FUSION REACTOR
                   Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTIONS FOR TOTAL NEUTRON EMISSIONS REQUIRED.
                   O: FOR NEUTRON MULTIPLICATION CALCULATION OF FUSION BLANKET.

63 041 NIOBIUM 000       NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  801110              801225
          1.00 EV    20.00 MEV    20.0%   2    KTO   K.SHIN
       800010           FUSION REACTOR
                   Q: LARGE DIFFERENCES BETWEEN EXPERIMENTAL DATA MEASURED AT ORNL/LASL AND KYOTO
                      UNIVERSITY.
                   O: MORE EXPERIMENTS WANTED.

64 041 NIOBIUM 093       NEUTRON                INELASTIC CROSS SECTION
   00  760401              761010
            UP TO    15.00 MEV    20.0%   2    MAP   M.KASAI
       780089  762117   FUSION REACTOR
                   Q: NB-93M PRODUCTION CROSS SECTION BY INELASTIC.
                   A: 15.0% REQUIRED FOR NEUTRON TRANSPORT CALCULATIONS.
                   O: TRANSMUTATION AND NEUTRON TRANSPORT CALCULATIONS.

65 041 NIOBIUM 093       NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   00  760401              761010
        0.25-1 EV    15.00 MEV    15.0%   2    JAE   Y.SEKI
       780092  762124   FUSION REACTOR
                   Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                   O: GAMMA-RAY HEATING CALCULATIONS.

66 041 NIOBIUM 093       NEUTRON                N,P
   00  760401              761010
            UP TO    15.00 MEV    20.0%   2    MAP   M.KASAI              MAP   K.IOKI
       780094  762119   FUSION REACTOR
                   O: HYDROGEN ACCUMULATION CALCULATIONS.

67 041 NIOBIUM 093       NEUTRON                TOTAL ALPHA PRODUCTION CROSS SECTION
   00  760401              761010
            UP TO    15.00 MEV    15.0%   2    MAP   K.IOKI
       780096  762121   FUSION REACTOR
                   O: HELIUM ACCUMULATION CALCULATIONS.
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68 041 NIOBIUM 093        NEUTRON                INELASTIC CROSS SECTION
   00  801110               801225
         0.25-1 EV      20.00 MEV    10.0%   1    PNC   M.SASAKI            JAE   K.SAKURAI
       800012             FISSION REACTOR
                    Q: PRODUCTION FOR 13.6 YEAR ISOMER.
                    C: FOR NEUTRON DOSIMETRY.


69 042 MOLYBDENUM         NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
   01  760401    801110    761010
          1.00 MEV     15.00 MEV    10.0%   2    JAE   Y.SEKI
       780099  762126    FUSION REACTOR
                    O: NEUTRON TRANSPORT CALCULATIONS.


70 042 MOLYBDENUM         NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
   01  760401    801110    761010
         0.25-1 EV      15.00 MEV    15.0%   2    JAE   Y.SEKI              MAP   K.IOKI
       780102  762131    FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: NEUTRON BALANCE AND GAMMA-RAY HEATING CALCULATION.


71 042 MOLYBDENUM         NEUTRON                N,P
   00  760401               761010
             UP TO     15.00 MEV    10.0%   2    JAE   Y.SEKI              MAP   K.IOKI
                                                 JAE   H.IIDA
       780104  762129    FUSION REACTOR
                    Q: CROSS SECTION FOR EACH ISOTOPE ARE ALSO REQUESTED. ESPECIALLY, DATA OF MO-95,-96
                       ARE REQUIRED FOR ESTIMATION OF DOSE RATES AROUND THE MOLYBDENUM STRUCTURES.
                    O: HYDROGEN ACCUMULATION CALCULATIONS AND FOR CALCULATION OF INDUCED ACTIVITIES.


72 042 MOLYBDENUM         NEUTRON                N,ALPHA
   00  760401               761010
             UP TO     15.00 MEV    20.0%   2    JAE   Y.SEKI              MAP   K.IOKI
       780105  762130    FUSION REACTOR
                    Q: CROSS SECTIONS FOR EACH ISOTOPE ARE ALSO REQUESTED.
                    O: HELIUM ACCUMULATION CALCULATIONS.


73 042 MOLYBDENUM 092     NEUTRON                N,NP
   00  780805               781212
             UP TO     15.00 MEV    20.0%   2    JAE   H.IIDA
       780108  792078    FUSION REACTOR
                    Q: EXPERIMENTAL DATA REQUIRED.
                    O: FOR CALCULATION OF INDUCED ACTIVITIES AROUND MOLYBDENUM STRUCTURES.


74 042 MOLYBDENUM 094     NEUTRON                N,2N
   00  760401               761010
             UP TO     15.00 MEV    10.0%   2    MAP   K.IOKI
       780111  762133    FISSION REACTOR
                    O: NEUTRON BALANCE AND TRANSMUTATION CALCULATIONS.
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75 043 TECHNETIUM 099    NEUTRON                CAPTURE CROSS SECTION
   01  750401    801110    751010
         25.00 KEV    25.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780124  752007   FISSION REACTOR
                   Q: ONE POINT ABSOLUTE DATA AT 25 KEV IS REQUIRED.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.

76 044 RUTHENIUM 102     NEUTRON                RESONANCE PARAMETERS
   00  801110              801225
             UP TO     3.00 KEV    20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800028           FISSION REACTOR
                   Q: ONLY 3 LEVELS ARE KNOWN UP TO 1.3 KEV.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.

77 044 RUTHENIUM 103                            GAMMA RAY YIELD
   00  720401              721010
                                     1.0%   2    JAE   K.TASAKA
       780126  722002   SAFEGUARDS
                   Q: YIELDS PER DISINTEGRATION OF 497 AND 610 KEV GAMMA-RAY REQUIRED.  (FOLLOWING
                      BETA DECAY EVENT).
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

78 044 RUTHENIUM 103     NEUTRON                CAPTURE CROSS SECTION
   01  780808    801110    781212
         100.00 EV    500.00 KEV    20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780127  792079   FISSION REACTOR
                   Q: EXPERIMENTAL DATA REQUIRED.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS. NO DIFFERENTIAL OR INTEGRAL DATA EXIST.
                      VERY LARGE DISCREPANCIES BETWEEN EVALUATIONS.

79 044 RUTHENIUM 104     NEUTRON                RESONANCE PARAMETERS
   00  801110              801225
             UP TO     3.00 KEV    20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800029           FISSION REACTOR
                   Q: ONLY 4 LEVELS ARE KNOWN UP TO 1.06 KEV.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.

80 046 PALLADIUM 105     NEUTRON                RESONANCE PARAMETERS
   00  801110              801225
           2.00 KEV     3.00 KEV    10.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800030           FISSION REACTOR
                   Q: STAVELOZ ET AL.(1979) MEASURED RESONANCE PARAMETERS BELOW 2 KEV.  ABOVE 3 KEV,
                      CAPTURE DATA OF MACKLIN ET AL. AGREE WITH THOSE BY HOCKENBURY ET AL.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.
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                    ** REQUESTS REGISTERED TO WRENDA.        **                          PAGE   13

81 046 PALLADIUM 107     NEUTRON                CAPTURE CROSS SECTION
   01  750401    801110    751010
         500.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780130  752012   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.  EVALUATIONS ARE VERY DISCREPANT.  NO KEV
                      DATA.

82 049 INDIUM 115        GAMMA                  SPECIAL QUANTITY
   00  780808              781212
         500.00 KEV    10.00 MEV    20.0%   3    TOK   Y.OKA
       780132  792080   FISSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED FOR (G,G') REACTION.
                   O: FOR CORRECTION OF IN-115M PRODUCTION THROUGH IN-115(N,N') IN-115M, FOR REACTOR
                      SHIELDING AND DOSIMETRY APPLICATIONS.

83 051 ANTIMONY 124      NEUTRON                CAPTURE CROSS SECTION
   00  780808              781212
         0.25-1 EV    0.25-1 EV     20.0%   3    JAE   K.NISHIMURA
       780135  792082   FISSION REACTOR
                   Q: EXPERIMENTAL DATA REQUIRED.
                   O: FOR ESTIMATION OF SB-124 PRODUCTION IN SB-BE NEUTRON SOURCE.  VERY LARG
                      DISCREPANCIES EXIST AMONG EXPERIMENTAL DATA.

84 053 IODINE 129        NEUTRON                CAPTURE CROSS SECTION
   00  801110              801225
         100.00 EV    500.00 KEV    20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800016           FISSION REACTOR
                   Q: NO EXPERIMENTAL DATA.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.

85 053 IODINE 129        NEUTRON                RESONANCE PARAMETERS
   00  801110              801225
                                   20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800017           FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.

86 053 IODINE 135                               GAMMA RAY YIELD
   00  760401              761010
                                   10.0%   3    TOS   H.SHIMOJIMA
       780137  762004   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 527,1132,1260, AND 1458 KEV GAMMA-RAYS REQUIRED.
                      (FOLLOWING BETA DECAY EVENT).
                   O: DETECTION OF FAILED FUEL.
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87 053 IODINE 137                                GAMMA RAY YIELD
   00  760401              761010
                                   10.0%   3    TOS   H.SHIMOJIMA
       780138  762005   SAFEGUARDS
                   Q: YIELD PER DESINTEGRATION OF MAJOR GAMMA-RAYS REQUIRED. (FOLLOWING BETA DECAY
                      EVENT).
                   O: DETECTION OF FAILED FUEL.

88 053 IODINE 138                                GAMMA RAY YIELD
   00  760401              761010
                                   10.0%   3    TOS   H.SHIMOJIMA
       780139  762006   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 589 KEV GAMMA-RAY REQUIRED. (FOLLOWING BETA DECAY
                      EVENT).
                   O: DETECTION OF FAILED FUEL.

89 053 IODINE 139                                HALF LIFE
   00  760401              761010
                                   10.0%   3    TOS   H.SHIMOJIMA
       780140  762013   SAFEGUARDS
                   O: DETECTION OF FAILED FUEL.

90 053 IODINE 139                                GAMMA RAY YIELD
   00  760401              761010
                                   10.0%   3    TOS   H.SHIMOJIMA
       780141  762007   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF MAJOR GAMMA-RAYS REQUIRED. (FOLLOWING BETA DECAY
                      EVENT).
                   O: DETECTION OF FAILE FUEL.

91 054 XENON 131         NEUTRON                 CAPTURE CROSS SECTION
   01  750401    801110    751010
         100.00 EV    500.00 KEV    20.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780142  752014   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS. NO EXPERIMENTAL DATA IN KEV REGION.
                      RESONANCE PARAMETERS ARE KNOWN UP TO 4 KEV.

92 054 XENON 132         NEUTRON                 CAPTURE CROSS SECTION
   00  801110              801225
         100.00 EV    500.00 KEV    20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800018           FISSION REACTOR
                   Q: NO EXPERIMENTAL DATA.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.

93 054 XENON 132         NEUTRON                 RESONANCE PARAMETERS
   00  801110              801225
                                   20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800019           FISSION REACTOR
                   Q: ONLY 3 LEVELS BELOW 4.0 KEV.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.
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94 054 XENON 139                                GAMMA RAY YIELD
   00  760401              761010
                                      10.0%   3    TOS   H.SHIMOJIMA
       780144  762008   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 175,219,290,297 AND 393 KEV GAMMA-RAYS REQUIRED.
                      (FOLLOWING BETA DECAY EVENT).
                   O: DETECTION OF FAILED FUEL.

95 055 CESIUM 134        NEUTRON                CAPTURE CROSS SECTION
   00  720401              721010
         0.25-1 EV    0.25-1 EV      3.0%   1    JAE   H.OKASHITA
       780148  722022   SAFEGUARDS
                   Q: RESONANCE INTEGRAL ALSO WANTED.
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

96 055 CESIUM 134        NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010
         0.25-1 EV     10.00 MEV    20.0%   1    JAE   K.TASAKA
       780149  762024   SAFEGUARDS
                   Q: CROSS-SECTION VALUES AT HIGHER NEUTRON ENERGIES ARE NEEDED, AS WELL AS AT
                      THERMAL ENERGY.
                   A: 10 PER CENT ACCURACY FOR 25.3 MV, 20 PER CENT ACCURACY FOR HIGHER ENERGY REGION.
                   O: BURNUP DETERMINATION BASED ON ABSOLUTE MEASUREMENT OF ACTIVITY RATIO
                      CS-134/CS-137 ESTIMATION OF THE DECAY POWER OF FISSION PRODUCTS.

97 055 CESIUM 135        NEUTRON                CAPTURE CROSS SECTION
   00  750401              751010
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780150  752016   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.  EVALUATIONS ARE VERY DISCREPANT.  NO DATA
                      AT ALL.  NO EXPERIMENTAL DATA FROM 100 EV TO 400 KEV.

98 055 CESIUM 135        NEUTRON                RESONANCE PARAMETERS
   00  801110              801225
                                      10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800020           FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.

99 061 PROMETHIUM 147    NEUTRON                CAPTURE CROSS SECTION
   01  750401    801110    751010
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780158  752019   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.
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100 062 SAMARIUM 149      NEUTRON                CAPTURE CROSS SECTION
    01  750401    801110    751010
          25.00 KEV    25.00 KEV     5.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
        780159  752020   FISSION REACTOR
                    Q: ONE POINT PRECISE MEASUREMENT AT 25 KEV IS REQUIRED.
                    O: FOR FAST REACTOR BURNUP CALCULATIONS.  DISCREPANCY BETWEEN STEK DATA AND RECENT
                       DIFFERENTIAL DATA.

101 062 SAMARIUM 151      NEUTRON                CAPTURE CROSS SECTION
    00  750401              751010
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
        780160  752021   FISSION REACTOR
                    O: FOR FAST REACTOR BURNUP CALCULATIONS.  NO KEV DATA.

102 063 EUROPIUM 152      NEUTRON                CAPTURE CROSS SECTION
    00  801110              801225
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
        800021           FISSION REACTOR
                    Q: NO KEV DATA.
                    O: FOR CONTROL ROD AND THERMAL REACTOR BURNUP CALCULATIONS.

103 063 EUROPIUM 152      NEUTRON                RESONANCE PARAMETERS
    00  801110              801225
                                      10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
        800022           FISSION REACTOR
                    Q: THERE EXIST NO DATA, EXCEPT THOSE BY VERTEBNYJ ET AL.(19 77) IN 0.88 TO 17 EV.
                    O: FOR CONTROL ROD AND THERMAL REACTOR BURNUP CALCULATIONS.

104 063 EUROPIUM 154      NEUTRON                CAPTURE CROSS SECTION
    00  720401              721010
          0.25-1 EV    0.25-1 EV      5.0%   1    JAE   H.OKASHITA
        780162  722039   SAFEGUARDS
                    O: RESONANCE INTEGRAL ALSO WANTED.
                    O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

105 063 EUROPIUM 154      NEUTRON                CAPTURE CROSS SECTION
    00  801110              801225
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
        800023           FISSION REACTOR
                    Q: NO EXPERIMENTAL DATA.
                    O: FOR CONTROL ROD AND THERMAL REACTOR BURNUP CALCULATIONS.

106 063 EUROPIUM 154      NEUTRON                RESONANCE PARAMETERS
    00  801110              801225
                                      10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
        800024           FISSION REACTOR
                    Q: INSUFFICIENT RESONANCE PARAMETERS.
                    O: FOR CONTROL ROD AND THERMAL REACTOR BURNUP CALCULATIONS.
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107 063 EUROPIUM 155                                GAMMA RAY YIELD
    00  720401              721010
                                        1.0%   2    JAE   K.TASAKA
        780163  722015   SAFEGUARDS
                    Q: YIELD PER DISINTEGRATION OF 86.5 AND 105.3 KEV GAMMA-RAYS REQUIRED.  (FOLLOWING
                       BETA DECAY EVENT).
                    O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

108 063 EUROPIUM 155      NEUTRON                   CAPTURE CROSS SECTION
    00  801110              801225
          100.00 EV    500.00 KEV    20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
        800025            FISSION REACTOR
                    Q: NO EXPERIMENTAL DATA.
                    O: FOR FAST REACTOR BURNUP CALCULATIONS.

109 063 EUROPIUM 155      NEUTRON                   RESONANCE PARAMETERS
    00  801110              801225
                                       20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
        800026            FISSION REACTOR
                    Q: INSUFFICIENT RESONANCE PARAMETERS.
                    O: FOR FAST REACTOR BURNUP CALCULATIONS.

110 073 TANTALUM 182      NEUTRON                   CAPTURE CROSS SECTION
    00  780808              781212
          0.25-1 EV    0.25-1 EV     10.0%   3    KTO   M.KOYAMA
        780164  792084   FISSION REACTOR
                    Q: EXPERIMENTAL DATA REQUIRED.
                    O: FOR ESTIMATION OF NEUTRON FLUENCE AND SPECTRUM.

111 079 GOLD 198          NEUTRON                   CAPTURE CROSS SECTION
    00  780808              781212
          0.25-1 EV    0.25-1 EV     10.0%   3    KTO   M.KOYAMA
        780165  792085   FISSION REACTOR
                    Q: EXPERIMENTAL DATA REQUIRED.
                    O: FOR ESTIMATION OF NEUTRON FLUENCE AND SPECTRUM.

112 080 MERCURY 199       NEUTRON                   INELASTIC CROSS SECTION
    00  801110              801225
          500.00 KEV    14.00 MEV    10.0%   3    JAE   K.SAKURAI
        800013            FISSION REACTOR
                    Q: PRODUCTION CROSS SECTION FOR 42.6 MIN ISOMER THROUGH INELASTIC SCATTERING.
                    O: FOR NEUTRON DOSIMETRY.
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113 082 LEAD              NEUTRON                TOTAL PHOTON PRODUCTION CROSS SECTION
    00  760401              761010
          0.25-1 EV     15.00 MEV    15.0%   2    JAE   Y.SEKI
        780166  762134   FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    A: AN UPPER LIMIT OF THE CROSS SECTION OR ACCURACY 20 PER CENT USEFUL.  NEUTRON
                       ENERGY RESOLUTION 300 KEV ABOVE 100 KEV AND 10 PER CENT OTHERWISE.  GAMMA ENERGY
                       RESOLUTION 1 MEV.
                    O: SHIELDING DESIGN AND GAMMA-RAY HEATING CALCULATION.

114 082 LEAD 000          NEUTRON                DOUBLE DIFFERENTIAL NEUTRON-EMISSION CROSS SECTION
    00  801110              801225
              UP TO     15.00 MEV    10.0%   2    OSA   A.TAKAHASHI
        800008           FUSION REACTOR
                    Q: ENERGY-ANGLE DIFFERENTIAL CROSS SECTIONS FOR TOTAL NEUTRON EMISSIONS REQUIRED.
                    O: FOR NEUTRON MULTIPLICATION CALCULATION OF FUSION BLANKET.

115 082 LEAD 206          NEUTRON                N,ALPHA
    00  780808              781212
              UP TO     15.00 MEV    20.0%   2    JAE   H.IIDA
        780167  792091   FUSION REACTOR
                    Q: EXPERIMENTAL DATA REQUIRED.
                    O: FOR FUSION REACTOR SHIELDING CALCULATION. FOR CALCULATION OF RESIDUAL ACTIVITY.
                       NO EXPERIMENTAL DATA EXCEPT FOR A FEW AT 14 MEV.

116 091 PROTACTINIUM 233  NEUTRON                CAPTURE CROSS SECTION
    00  760401              761010
          20.00 EV      15.00 MEV    10.0%   1    JAE   R.SHINDO
        780168  762208   FISSION REACTOR
                    O: FOR BURNUP CALCULATION OF THORIUM FUELED THERMAL REACTORS.

117 092 URANIUM 233       NEUTRON                CAPTURE CROSS SECTION
    00  780808              781212
           1.00 MEV     20.00 MEV    10.0%   1    SAE   N.ASANO
        780169  792083   FISSION REACTOR
                    Q: EXPERIMENTAL DATA REQUIRED.

118 092 URANIUM 233       NEUTRON                N,2N
    00  780808              781212
              UP TO     20.00 MEV    10.0%   1    SAE   N.ASANO
        780170  792092   FISSION REACTOR
                    Q: EXPERIMENTAL DATA WANTED.

119 092 URANIUM 235       NEUTRON                CAPTURE CROSS SECTION
    00  680401              681010
           1.00 MEV     10.00 MEV            1    JAE   S.KATSURAGI         SAE   H.MATSUNOBU
        780173  682055   FISSION REACTOR
                    Q: ALPHA ALSO WANTED.
                    A: REQUIRED ACCURACY - 5 TO 10 PER CENT.  RESOLUTION - 1 TO 2 PER CENT.
                    O: FOR FAST REACTORS.  NUCLEAR DATA EVALUATION. NO EXPERIMENTAL DATA ABOVE 2.6 MEV.
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120 092 URANIUM 235       NEUTRON                DELAYED NEUTRONS EMITTED PER FISSION
    00  760401              761010
          0.25-1 EV     10.00 MEV     5.0%   2    NIG   T.MURATA
        780174  762046   SAFEGUARDS
                    Q: THE REQUESTED QUANTITIES ARE THE GROUP HALF LIVES AND GROUP YIELDS (NORMALIZED
                       TO 1 FISSION) WHICH CAN BE USED TO FIT THE DECAY CURVE OF DELAYED NEUTRONS FOR
                       THE TIME RANGE 0.1-300 SEC WITHIN AN ACCURACY OF 5 PER CENT.
                    O: INCIDENT ENERGY STEP LESS THAN 2 MEV.  ACTIVE ASSAY OF MIXED FRESH AND
                       IRRADIATED FUEL.

121 092 URANIUM 236       NEUTRON                CAPTURE CROSS SECTION
    00  720401              721010
          0.25-1 EV     14.00 MEV            2    JAE   Y.NAITO
        780175  722040   SAFEGUARDS
                    A: ACCURACY REQUIRED AT THERMAL IS 3 PER CENT, 10 PER CENT ABOVE.
                    O: FOR BURNUP CALCULATION OF A PU LOADED THERMAL REACTOR.

122 092 URANIUM 237                              GAMMA RAY YIELD
    00  780808              781212
                                       5.0%   2    NIS   Y.NODA              NIS   H.OKABAYASHI
        780176  792090   FISSION REACTOR
                    Q: YIELD PER DISINTEGRATION OF 59.5 AND 208 KEV GAMMA RAYS.
                    O: RADIATION DOSE CALCULATION FOR PU-241 DAUGHTER. STATUS NUCLEAR DATA SHEETS, 23
                       P1 (1978) 2 EVALUATION 10 M.

123 092 URANIUM 238       NEUTRON                DELAYED NEUTRONS EMITTED PER FISSION
    00  760401              761010
          0.25-1 EV     10.00 MEV     5.0%   2    NIG   T.MURATA
        780179  762047   SAFEGUARDS
                    Q: THE REQUESTED QUANTITIES ARE THE GROUP HALF LIVES AND GROUP YIELDS (NORMALIZED
                       TO 1 FISSION) WHICH CAN BE USED TO FIT THE DECAY CURVE OF DELAYED NEUTRONS FOR
                       THE TIME RANGE 0.1 - 300 SEC WITHIN AN ACCURACY OF 5.0 PER CENT.
                    O: INCIDENT ENERGY STEP LESS THAN 2 MEV.  ACTIVE ASSAY OF MIXED FRESH AND
                       IRRADIATED FUEL.

124 092 URANIUM 238       NEUTRON                FISSION PRODUCT MASS YIELD SPECTRUM
    00  760401              761010
                                      10.0%   3    TOS   H.SHIMOJIMA
        780180  762044   SAFEGUARDS
                    Q: CUMULATIVE YIELDS OF BR-87,BR-88,KR-90,I-137,I-138,I-139, XE-137,XE-138 FOR
                       FISSION NEUTRON AND 1 - 14 MEV NEUTRON SPECTRA.
                    O: DETECTION OF FAILED FUEL.

125 093 NEPTUNIUM 237     NEUTRON                CAPTURE CROSS SECTION
    00  780808              781212
          0.25-1 EV      1.00 KEV    10.0%   1    PNC   I.OHTAKE
        780181  792086   FISSION REACTOR
                    Q: EXPERIMENTAL DATA WANTED.  EVALUATION DESIRABLE. RESONANCE PARAMETERS ARE ALSO
                       REQUIRED.
                    O: FOR BURNUP CALCULATION OF THERMAL AND FAST REACTORS.
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126 093 NEPTUNIUM 237     NEUTRON                CAPTURE CROSS SECTION
    00  780808              781212
            1.00 KEV    15.00 MEV    20.0%   1    PNC   I.OHTAKE
        780182  792089   FISSION REACTOR
                    Q: EXPERIMENTAL DATA REQUIRED.  EVALUATION DESIRABLE.
                    O: FOR BURNUP CALCULATION OF THERMAL AND FAST REACTORS.

127 093 NEPTUNIUM 237     NEUTRON                FISSION CROSS SECTION
    00  760401              761010
                UP TO   15.00 MEV     1.0%   2    JAE   Y.SEKI
        780183  762135   FUSION REACTOR
                    Q: RATIO TO U-235 FISSION USEFUL.
                    A: ACCURACY 3 PER CENT USEFUL.  NEUTRON ENERGY RESOLUTION 300 KEV.
                    O: FOR MONITOR REACTION AND RADIATION DOSIMETRY IN NEUTRONICS EXPERIMENTS ON
                       BLANKET SYSTEM OF FUSION REACTORS.

128 093 NEPTUNIUM 239     NEUTRON                CAPTURE CROSS SECTION
    00  710401              711010
           10.00 KEV     5.00 MEV    20.0%   3    KYU   M.OHTA
        780184  712075   FISSION REACTOR
                    Q: SOME POINT DATA ARE ALSO USEFUL.
                    O: FOR NORMALIZATION OF CALCULATED CAPTURE CROSS SECTION. FOR BURNUP CALCULATION.

129 093 NEPTUNIUM 239     NEUTRON                CAPTURE CROSS SECTION
    00  760401              761010
          0.25-1 EV     15.00 MEV    20.0%   2    JAE   R.SHINDO
        780186  762209   FISSION REACTOR
                    O: FOR BURNUP CALCULATION OF THERMAL REACTOR.

130 094 PLUTONIUM 238     SPONTANEOUS            FISSION HALF LIFE
    01  760401    801110    761010
                                      1.0%   2    PNC   S.SUZUKI
        780188  762014   SAFEGUARDS
                    O: DETECTION OF PU BY NEUTRON COINCIDENCE METHOD.

131 094 PLUTONIUM 238                            GAMMA RAY YIELD
    00  760401              761010
                                      1.0%   1    JAE   T.SUZUKI
        780189  762009   SAFEGUARDS
                    Q: YIELD PER DISINTEGRATION OF 43.45,99.7,152.7 KEV GAMMA RAYS REQUIRED.
                       (FOLLOWING ALPHA DECAY EVENT).
                    O: THOUGH PRESENT STATUS OF ACCURACY SEEMED TO MEET THE REQUIREMENT CONFIRMATION IS
                       REQUIRED.  ASSAY OF PU ISOTOPES BY GAMMA-RAY SPECTROSCOPY.

132 094 PLUTONIUM 238     NEUTRON                CAPTURE CROSS SECTION
    00  780808              781212
          0.25-1 EV    500.00 KEV    20.0%   2    PNC   I.OHTAKE
        780191  792087   FISSION REACTOR
                    Q: EXPERIMENTAL DATA DESIRED.  EVALUATED DATA ALSO REQUIRED.
                    O: FOR BURNUP CALCULATION OF THERMAL AND FAST REACTORS.
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133 094 PLUTONIUM 238     NEUTRON                CAPTURE CROSS SECTION
    00  780808              781212
          500.00 KEV    15.00 MEV    10.0%   2    PNC   I.OHTAKE
        780192  792088   FISSION REACTOR
                    Q: EXPERIMENTAL DATA DESIRED.  EVALUATED DATA ALSO REQUIRED.
                    O: FOR BURNUP CALCULATION OF THERMAL AND FAST REACTORS.

134 094 PLUTONIUM 238                            SPECIAL QUANTITY
    01  760401    801110    761010
                                     0.5%   1    PNC   S.SUZUKI
        780193  762018   SAFEGUARDS
                    Q: DECAY HEAT (W/G) REQUIRED.
                    O: ASSAY OF PU BY CALORIMETRY.

135 094 PLUTONIUM 239                            GAMMA RAY YIELD
    00  760401              761010
                                     1.0%   1    JAE   T.SUZUKI
        780195  762010   SAFEGUARDS
                    Q: YIELD PER DISINTEGRATION OF 45.2,104.2 AND 642.3 KEV GAMMA RAYS REQUIRED.
                       (FOLLOWING ALPHA DECAY EVENT).
                    O: THOUGH PRESENT STATUS OF ACCURACY SEEMED TO MEET THE REQUIREMENT CONFIRMATION IS
                       REQUIRED.  ASSAY OF PU ISOTOPES BY GAMMA-RAY SPECTROSCOPY.

136 094 PLUTONIUM 239     NEUTRON                TOTAL CROSS SECTION
    01  760401    801110    761010
            1.00 KEV   200.00 KEV     2.0%   1    NIG   M.KAWAI
        780198  762210   FISSION REACTOR
                    O: FISSION REACTOR CALCULATIONS.

137 094 PLUTONIUM 239     NEUTRON                FISSION CROSS SECTION
    01  760401    801110    761010
           10.00 KEV    20.00 MEV     3.0%   1    NIG   M.KAWAI
        780199  762211   FISSION REACTOR
                    O: FISSION REACTOR CORE DESIGN AND ANALYSIS.  LARGE DISCREPANCY BETWEEN
                       EXPERIMENTAL DATA FROM 50 KEV TO 1.0 MEV.

138 094 PLUTONIUM 239     NEUTRON                CAPTURE TO FISSION RATIO(ALPHA)
    00  720401              721010
          0.25-1 EV     14.00 MEV            2    JAE   Y.NAITO
        780200  722046   SAFEGUARDS
                    A: ACCURACY REQUIRED AT THERMAL IS 1 PER CENT, 5 PER CENT ABOVE.
                    O: FOR BURNUP CALCULATION OF A PU LOADED THERMAL REACTOR.

139 094 PLUTONIUM 239     NEUTRON                NEUTRONS EMITTED PER FISSION(NU BAR)
    01  700401    801110    701010
              UP TO    15.00 MEV     0.5%   1    NIG   M.KAWAI
        780201  702037   FISSION REACTOR
                    A: ACCURAY REQUIRED TO BETTER THAN 0.2 PER CENT, IF POSSIBLE.
                    O: FOR FAST AND HYBRID FUSION REACTOR CALCULATIONS. DISCREPANCY EXISTS BETWEEN
                       EXPERIMENTAL DATA.
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140 094 PLUTONIUM 239     NEUTRON                NEUTRONS EMITTED PER FISSION(NU BAR)
    00  720401              721010
          0.25-1 EV    0.25-1 EV      0.5%   2    JAE   Y.NAITO
        780202  722048   SAFEGUARDS
                    Q: DATA WANTED FOR EPI-THERMAL NEUTRONS ALSO.
                    O: FOR BURNUP CALCULATION OF A PU LOADED THERMAL REACTOR.

141 094 PLUTONIUM 239     NEUTRON                DELAYED NEUTRONS EMITTED PER FISSION
    00  760401              761010
          0.25-1 EV     10.00 MEV     5.0%   2    NIG   T.MURATA
        780203  762048   SAFEGUARDS
                    Q: THE REQUESTED QUANTITIES ARE THE GROUP HALF LIVES AND GROUP YIELDS (NORMALIZED
                       TO 1 FISSION) WHICH CAN BE USED TO FIT THE DECAY CURVE OF DELAYED NEUTRONS FOR
                       THE TIME RANGE 0.1 - 300 SEC WITHIN AN ACCURACY OF 5 PER CENT.
                    O: INCIDENT ENERGY STEP LESS THAN 2 MEV.  NOTIVE ASSAY OF MIXED FRESH AND
                       IRRADIATED FUEL.

142 094 PLUTONIUM 239                            SPECIAL QUANTITY
    01  760401    801110    761010
                                       0.5%   1    PNC   S.SUZUKI
        780204  762019   SAFEGUARDS
                    Q: DECAY HEAT (W/G) REQUIRED.
                    O: ASSAY OF PU BY CALORIMETRY.

143 094 PLUTONIUM 239     NEUTRON                CAPTURE TO FISSION RATIO(ALPHA)
    00  801110              801225
            1.00 MEV    20.00 MEV    10.0%   2    PNC   M.SASAKI
        800015            FISSION REACTOR
                    Q: INSUFFICIENT EXPERIMENTAL DATA.
                    O: FOR CALCULATION OF FBR BREEDING RATIO.  EVALUATED DATA WANTED.

144 094 PLUTONIUM 240     SPONTANEOUS            FISSION HALF LIFE
    01  760401    801110    761010
                                       1.0%   2    PNC   S.SUZUKI
        780205  762016   SAFEGUARDS
                    O: DETECTION OF PU BY NEUTRON COINCIDENCE METHOD.

145 094 PLUTONIUM 240                            GAMMA RAY YIELD
    00  760401              761010
                                       1.0%   1    JAE   T.SUZUKI
        780206  762011   SAFEGUARDS
                    Q: YIELD PER DISINTEGRATION OF 45.2,104.2 AND 642.3 KEV GAMMA-RAYS REQUIRED.
                       (FOLLOWING ALPHA DECAY EVENT).
                    O: THOUGH PRESENT STATUS OF ACCURACY SEEMED TO MEET THE REQUIREMENT CONFIRMATION IS
                       REQUIRED.  ASSAY OF PU ISOTOPES BY GAMMA-RAY SPECTROSCOPY.
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146 094 PLUTONIUM 240     NEUTRON                FISSION CROSS SECTION
    01  760401    801110    761010
          0.25-1 EV      1.00 MEV    10.0%   1    PNC   M.SASAKI
        780209  762213   FISSION REACTOR
                    Q: FOR FAST REACTOR CALCULATIONS.

147 094 PLUTONIUM 240     NEUTRON                DELAYED NEUTRONS EMITTED PER FISSION
    00  760401              761010
          0.25-1 EV     10.00 MEV     5.0%   2    NIG   T.MURATA
        780210  762049   SAFEGUARDS
                    Q: THE REQUESTED QUANTITIES ARE THE GROUP HALF LIVES AND GROUP YIELDS (NORMALIZED
                       TO 1 FISSION) WHICH CAN BE USED TO FIT THE DECAY CURVE OF DELAYED NEUTRONS FOR
                       THE TIME RANGE 0.1-300 SEC WITHIN AN ACCURACY OF 5 PER CENT.
                    O: INCIDENT ENERGY STEP LESS THAN 2 MEV.  ACTIVE ASSAY OF MIXED FRESH AND
                       IRRADIATED FUEL.

148 094 PLUTONIUM 240     NEUTRON                RESONANCE PARAMETERS
    01  760401    801110    761010
          1.00 EV     10.00 KEV            1    PNC   M.SASAKI
        780211  762215   FISSION REACTOR
                    O: FOR FAST REACTOR CALCULATIONS.

149 094 PLUTONIUM 240                            SPECIAL QUANTITY
    01  760401    801110    761010
                                     0.5%   1    PNC   S.SUZUKI
        780212  762020   SAFEGUARDS
                    Q: DECAY HEAT (W/G) REQUIRED.
                    O: ASSAY OF PU BY CALORIMETRY.

150 094 PLUTONIUM 241     NEUTRON                GAMMA RAY YIELD
    00  760401              761010
                                     5.0%   1    JAE   T.SUZUKI
        780213  762012   SAFEGUARDS
                    Q: YIELD PER DISINTEGRATION OF 56.4,77,103.5,148.6 AND 160 KEV GAMMA-RAYS REQUIRED.
                       (FOLLOWING ALPHA DECAY EVENT).
                    A: 1 PER CENT ACCURACY FOR 103.5 AND 148.6 KEV GAMMA RAYS, 5 PER CENT ACCURACY FOR
                       56.4,77 AND 160 KEV GAMMA RAYS.
                    O: THOUGH PRESENT STATUS OF ACCURACY SEEMED TO MEET THE REQUIREMENT CONFIRMATION IS
                       REQUIRED.  ASSAY OF PU ISOTOPES BY GAMMA-RAY SPECTROSCOPY.

151 094 PLUTONIUM 241     NEUTRON                CAPTURE TO FISSION RATIO(ALPHA)
    00  720401              721010
          0.25-1 EV     14.00 MEV            2    JAE   Y.NAITO
        780218  722047   SAFEGUARDS
                    A: ACCURACY REQUIRED AT THERMAL IS 1 PER CENT, 5 PER CENT ABOVE.
                    O: FOR BURNUP CALCULATION OF A PU LOADED THERMAL REACTOR.
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152 094 PLUTONIUM 241     NEUTRON                DELAYED NEUTRONS EMITTED PER FISSION
    00  760401              761010
          0.25-1 EV     10.00 MEV     5.0%   2    NIG   T.MURATA
        780220  762050   SAFEGUARDS
                    Q: THE REQUESTED QUANTITIES ARE THE GROUP HALF LIVES AND GROUP YIELDS (NORMALIZED
                       TO 1 FISSION) WHICH CAN BE USED TO FIT THE DECAY CURVE OF DELAYED NEUTRONS FOR
                       THE TIME RANGE 0.1 - 300 SEC WITHIN AN ACCURACY OF 5 PER CENT.
                    O: ACTIVE ASSAY OF MIXED FRESH AND IRRADIATED FUEL INCIDENT ENERGY STEP LESS THAN 2
                       MEV.

153 094 PLUTONIUM 241                            SPECIAL QUANTITY
    01  760401    801110    761010
                                        0.5%   1    PNC   S.SUZUKI
        780222  762021   SAFEGUARDS
                    Q: DECAY HEAT (W/G) REQUIRED.
                    O: ASSAY OF PU BY CALORIMETRY.

154 094 PLUTONIUM 242     SPONTANEOUS            FISSION HALF LIFE
    01  760401    801110    761010
                                        1.0%   2    PNC   S.SUZUKI
        780223  762017   SAFEGUARDS
                    O: DETECTION OF PU BY NEUTRON COINCIDENCE METHOD.

155 094 PLUTONIUM 242     NEUTRON                CAPTURE CROSS SECTION
    00  720401              721010
          0.25-1 EV     14.00 MEV            2    JAE   Y.NAITO
        780224  722043   SAFEGUARDS
                    A: ACCURACY REQUIRED AT THERMAL IS 5 PER CENT, 10 PER CENT ABOVE.
                    O: FOR BURNUP CALCULATION OF A PU LOADED THERMAL REACTOR.

156 094 PLUTONIUM 242                            SPECIAL QUANTITY
    01  760401    801110    761010
                                        0.5%   1    PNC   S.SUZUKI
        780227  762022   SAFEGUARDS
                    Q: DECAY HEAT (W/G) REQUIRED.
                    O: ASSAY OF PU BY CALORIMETRY.

157 095 AMERICIUM 241     NEUTRON                CAPTURE CROSS SECTION
    01  750401    801110    751010
          500.00 KEV    15.00 MEV    10.0%   1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
                                                   MAP   T.HOJUYAMA
        780230  752033   FISSION REACTOR
                    Q: PRODUCTION OF AM-242 AND AM-242M WANTED.
                    O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                       SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.
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158 095 AMERICIUM 241                          SPECIAL QUANTITY
    01  760401    801110    761010
                                     0.5%   1    PNC   S.SUZUKI
        780231  762023   SAFEGUARDS
                    Q: DECAY HEAT (W/G) REQUIRED.
                    O: ASSAY OF PU BY CALORIMETRY.

159 095 AMERICIUM 242     NEUTRON              CAPTURE CROSS SECTION
    00  760401              761010
          0.25-1 EV     10.00 MEV    10.0%   3    NFI   M.YADA
        780234  762027   SAFEGUARDS
                    O: FOR HIGHER BURNUP CALCULATIONS.

160 095 AMERICIUM 243     NEUTRON              CAPTURE CROSS SECTION
    00  760401              761010
          0.25-1 EV      2.00 MEV    20.0%   3    NFI   M.YADA              TIT   K.EBIZUKA
        780236  762028   SAFEGUARDS
                    Q: TOTAL,ELASTIC AND INELASTIC CROSS SECTIONS ARE ALSO REQUIRED BY K.EBIZUKA TIT.
                    A: 10 PER CENT ACCURACY FOR 25 MV, 20 PER CENT ACCURACY FOR HIGHER ENERGY REGION.
                    O: BURNUP ANALYSIS OF FAST BREEDER REACTORS.

161 095 AMERICIUM 243     NEUTRON              CAPTURE CROSS SECTION
    00  801110              801225
           20.00 EV     15.00 MEV            1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
                                                  JAE   R.SHINDO            MAP   T.HOJUYAMA
        800027            FISSION REACTOR
                    Q: CAPTURE CROSS SECTIONS TO GROUND AND ISOMER STATES OF AM-244 REQUIRED.  THE
                       EXPERIMENTAL DATA ARE VERY SCARCE IN KEV AND MEV REGIONS.
                    O: FOR BURNUP CALCULATIONS OF THERMAL AND FAST REACTORS, ESTIMATION FOR BUILD-UP OF
                       TRANSURANIUM NUCLIDES IN SPENT FUEL, AND NEUTRON SHIELDING DESIGN FOR TRANSPORT
                       CASK OF SPENT FUEL.

162 096 CURIUM 242        NEUTRON              CAPTURE CROSS SECTION
    00  750401              751010
          0.25-1 EV     15.00 MEV            1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
                                                  MAP   T.HOJUYAMA
        780238  752042   FISSION REACTOR
                    A: ACCURACY REQUIRED 10 TO 20 PER CENT.
                    O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                       SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.

163 096 CURIUM 242        NEUTRON              FISSION CROSS SECTION
    00  750401              751010
          0.25-1 EV     15.00 MEV            1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
                                                  MAP   T.HOJUYAMA
        780240  752041   FISSION REACTOR
                    A: ACCURACY REQUIRED 10 TO 20 PER CENT.
                    O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                       SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.
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164 096 CURIUM 243        NEUTRON                CAPTURE CROSS SECTION
    00  750401              751010
           20.00 EV     10.00 MEV            1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
        780241  752047   FISSION REACTOR
                    A: ACCURACY REQUIRED 10 TO 20 PER CENT.
                    O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                       SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.

165 096 CURIUM 243        NEUTRON                FISSION CROSS SECTION
    00  750401              751010
            3.00 MEV    10.00 MEV            1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
        780243  752045   FISSION REACTOR
                    A: ACCURACY REQUIRED 10 TO 20 PER CENT.
                    O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                       SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.
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3. High Priority Requests

Requests of priority 1 are presented in this section. Among these, 30 requests are submitted to NEANDC as the high priority requests. They are those having the following sequential number:

1, 2, 6, 7, 8, 11, 12, 14, 15, 16, 17, 18, 19, 20, 21,

22, 23, 25, 26, 27, 31, 32, 33, 36, 42, 44, 45, 46, 47, 48.

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                    ** SPECIAL RETRIEVAL FROM JAPANESE REQUEST. **                    PAGE    1

1 003 LITHIUM 006       NEUTRON                N,NT
  00  760401              761010
          3.00 MEV    15.00 MEV     5.0%   1    JAE   Y.SEKI
      780005  762053   FUSION REACTOR
                 O: TRITIUM BREEDING AND ENERGY DEPOSITION CALCULATION.

2 003 LITHIUM 007       NEUTRON                N,NT
  01  760401    801110    761010
            UP TO    15.00 MEV     5.0%   1    JAE   Y.SEKI              OSA   A.TAKAHASHI
      780013  762058   FUSION REACTOR
                 Q: NEUTRON SPECTRA WITH ACCURACY 15 PER CENT ALSO REQUIRED.
                 O: TRITIUM BREEDING AND ENERGY DEPOSITION CALCULATION.

3 026 IRON              NEUTRON                INELASTIC CROSS SECTION
  01  760401    801110    761010
            UP TO    20.00 MEV     5.0%   1    JAE   Y.SEKI              NIG   M.KAWAI
      780064  762099   FUSION REACTOR
                 Q: INELASTIC GAMMA RAY SPECTRA ALSO REQUIRED.
                 O: NEUTRON TRANSPORT AND GAMMA-RAY HEATING CALCULATIONS.

4 028 NICKEL            NEUTRON                INELASTIC CROSS SECTION
  01  760401    801110    761010
            UP TO    20.00 MEV     5.0%   1    JAI   Y.SEKI              MAP   M.KASAI
                                             NIG   M.KAWAI
      780072  762105   FUSION REACTOR
                 Q: INELASTIC GAMMA-RAY SPECTRA ALSO REQUIRED.
                 O: NEUTRON TRANSPORT AND GAMMA-RAY HEATING CALCULATIONS.

5 041 NIOBIUM 093       NEUTRON                INELASTIC CROSS SECTION
  00  801110              801225
        0.25-1 EV     20.00 MEV    10.0%   1    PNC   M.SASAKI            JAE   K.SAKURAI
      800012           FISSION REACTOR
                 Q: PRODUCTION FOR 13.6 YEAR ISOMER.
                 O: FOR NEUTRON DOSIMETRY.

6 043 TECHNETIUM 099    NEUTRON                CAPTURE CROSS SECTION
  01  750401    801110    751010
         25.00 KEV    25.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
      780124  752007   FISSION REACTOR
                 Q: ONE POINT ABSOLUTE DATA AT 25 KEV IS REQUIRED.
                 O: FOR FAST REACTOR BURNUP CALCULATIONS.

7 046 PALLADIUM 107     NEUTRON                CAPTURE CROSS SECTION
  01  750401    801110    751010
        500.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
      780130  752012   FISSION REACTOR
                 O: FOR FAST REACTOR BURNUP CALCULATIONS.  EVALUATIONS ARE VERY DISCREPANT.  NO KEV
                    DATA.
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 8 054 XENON 131          NEUTRON                CAPTURE CROSS SECTION
   01  750401    801110    751010
         100.00 EV    500.00 KEV    20.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780142  752014   FISSION REACTOR
                    O: FOR FAST REACTOR BURNUP CALCULATIONS. NO EXPERIMENTAL DATA IN KEV REGION.
                       RESONANCE PARAMETERS ARE KNOWN UP TO 4 KEV.

 9 055 CESIUM 134         NEUTRON                CAPTURE CROSS SECTION
   00  720401              721010
         0.25-1 EV    0.25-1 EV      3.0%   1    JAE   H.OKASHITA
       780148  722022   SAFEGUARDS
                    Q: RESONANCE INTEGRAL ALSO WANTED.
                    O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

10 055 CESIUM 134         NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010
         0.25-1 EV     10.00 MEV    20.0%   1    JAE   K.TASAKA
       780149  762024   SAFEGUARDS
                    Q: CROSS-SECTION VALUES AT HIGHER NEUTRON ENERGIES ARE NEEDED, AS WELL AS AT
                       THERMAL ENERGY.
                    A: 10 PER CENT ACCURACY FOR 25.3 MV, 20 PER CENT ACCURACY FOR HIGHER ENERGY REGION.
                    O: BURNUP DETERMINATION BASED ON ABSOLUTE MEASUREMENT OF ACTIVITY RATIO
                       CS-134/CS-137 ESTIMATION OF THE DECAY POWER OF FISSION PRODUCTS.

11 055 CESIUM 135         NEUTRON                CAPTURE CROSS SECTION
   00  750401              751010
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780150  752016   FISSION REACTOR
                    O: FOR FAST REACTOR BURNUP CALCULATIONS.  EVALUATIONS ARE VERY DISCREPANT.  NO DATA
                       AT ALL.  NO EXPERIMENTAL DATA FROM 100 EV TO 400 KEV.

12 055 CESIUM 135         NEUTRON                RESONANCE PARAMETERS
   00  801110              801225
                                     10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800020           FISSION REACTOR
                    O: FOR FAST REACTOR BURNUP CALCULATIONS.

13 059 PRASEODYMIUM 144                          GAMMA RAY YIELD
   00  720401              721010
                                      1.0%   1    JAE   H.OKASHITA
       780155  722012   SAFEGUARDS
                    Q: YIELD PER DISINTEGRATION OF 696.5,1498.1 AND 2185.7 KEV GAMMA-RAYS REQUIRED.
                       (FOLLOWING BETA DECAY EVENT).
                    O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.
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14 061 PROMETHIUM 147    NEUTRON                CAPTURE CROSS SECTION
   01  750401    801110    751010
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780158  752019   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.

15 062 SAMARIUM 149      NEUTRON                CAPTURE CROSS SECTION
   01  750401    801110    751010
          25.00 KEV    25.00 KEV     5.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780159  752020   FISSION REACTOR
                   Q: ONE POINT PRECISE MEASUREMENT AT 25 KEV IS REQUIRED.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.  DISCREPANCY BETWEEN STEK DATA AND RECENT
                      DIFFERENTIAL DATA.

16 062 SAMARIUM 151      NEUTRON                CAPTURE CROSS SECTION
   00  750401              751010
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780160  752021   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.  NO KEV DATA.

17 063 EUROPIUM 152      NEUTRON                CAPTURE CROSS SECTION
   00  801110              801225
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800021           FISSION REACTOR
                   Q: NO KEV DATA.
                   O: FOR CONTROL ROD AND THERMAL REACTOR BURNUP CALCULATIONS.

18 063 EUROPIUM 152      NEUTRON                RESONANCE PARAMETERS
   00  801110              801225
                                     10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800022           FISSION REACTOR
                   Q: THERE EXIST NO DATA, EXCEPT THOSE BY VERTEBNYJ ET AL.(19 77) IN 0.88 TO 17 EV.
                   O: FOR CONTROL ROD AND THERMAL REACTOR BURNUP CALCULATIONS.

19 063 EUROPIUM 154      NEUTRON                CAPTURE CROSS SECTION
   00  720401              721010
         0.25-1 EV    0.25-1 EV      5.0%   1    JAE   H.OKASHITA
       780162  722039   SAFEGUARDS
                   Q: RESONANCE INTEGRAL ALSO WANTED.
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

20 063 EUROPIUM 154      NEUTRON                CAPTURE CROSS SECTION
   00  801110              801225
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800023           FISSION REACTOR
                   Q: NO EXPERIMENTAL DATA.
                   O: FOR CONTROL ROD AND THERMAL REACTOR BURNUP CALCULATIONS.
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21 063 EUROPIUM 154      NEUTRON                RESONANCE PARAMETERS
   00  801110              801225
                                   10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       800024            FISSION REACTOR
                   Q: INSUFFICIENT RESONANCE PARAMETERS.
                   O: FOR CONTROL ROD AND THERMAL REACTOR BURNUP CALCULATIONS.

22 091 PROTACTINIUM 233  NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010
          20.00 EV     15.00 MEV    10.0%   1    JAE   R.SHINDO
       780168  762208   FISSION REACTOR
                   O: FOR BURNUP CALCULATION OF THORIUM FUELED THERMAL REACTORS.

23 092 URANIUM 233       NEUTRON                CAPTURE CROSS SECTION
   00  780808              781212
           1.00 MEV    20.00 MEV    10.0%   1    SAE   N.ASANO
       780169  792083   FISSION REACTOR
                   Q: EXPERIMENTAL DATA REQUIRED.

24 092 URANIUM 233       NEUTRON                N,2N
   00  780808              781212
              UP TO    20.00 MEV    10.0%   1    SAE   N.ASANO
       780170  792092   FISSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED.

25 092 URANIUM 235       NEUTRON                CAPTURE CROSS SECTION
   00  680401              681010
           1.00 MEV    10.00 MEV            1    JAE   S.KATSURAGI         SAE   H.MATSUNOBU
       780173  682055   FISSION REACTOR
                   Q: ALPHA ALSO WANTED.
                   A: REQUIRED ACCURACY - 5 TO 10 PER CENT.  RESOLUTION - 1 TO 2 PER CENT.
                   O: FOR FAST REACTORS.  NUCLEAR DATA EVALUATION. NO EXPERIMENTAL DATA ABOVE 2.6 MEV.

26 093 NEPTUNIUM 237     NEUTRON                CAPTURE CROSS SECTION
   00  780808              781212
         0.25-1 EV      1.00 KEV    10.0%   1    PNC   I.OHTAKE
       780181  792086   FISSION REACTOR
                   Q: EXPERIMENTAL DATA WANTED.  EVALUATION DESIRABLE. RESONANCE PARAMETERS ARE ALSO
                      REQUIRED.
                   O: FOR BURNUP CALCULATION OF THERMAL AND FAST REACTORS.

27 093 NEPTUNIUM 237     NEUTRON                CAPTURE CROSS SECTION
   00  780808              781212
           1.00 KEV    15.00 MEV    20.0%   1    PNC   I.OHTAKE
       780182  792089   FISSION REACTOR
                   Q: EXPERIMENTAL DATA REQUIRED.  EVALUATION DESIRABLE.
                   O: FOR BURNUP CALCULATION OF THERMAL AND FAST REACTORS.
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28 094 PLUTONIUM 238                                GAMMA RAY YIELD
   00  760401              761010
                                     1.0%   1    JAE   T.SUZUKI
       780189  762009   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 43.45,99.7,152.7 KEV GAMMA RAYS REQUIRED.
                      (FOLLOWING ALPHA DECAY EVENT).
                   O: THOUGH PRESENT STATUS OF ACCURACY SEEMED TO MEET THE REQUIREMENT CONFIRMATION IS
                      REQUIRED.  ASSAY OF PU ISOTOPES BY GAMMA-RAY SPECTROSCOPY.

29 094 PLUTONIUM 238                                SPECIAL QUANTITY
   01  760401    801110    761010
                                     0.5%   1    PNC   S.SUZUKI
       780193  762018   SAFEGUARDS
                   Q: DECAY HEAT (W/G) REQUIRED.
                   O: ASSAY OF PU BY CALORIMETRY.

30 094 PLUTONIUM 239                                GAMMA RAY YIELD
   00  760401              761010
                                     1.0%   1    JAE   T.SUZUKI
       780195  762010   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 45.2,104.2 AND 642.3 KEV GAMMA RAYS REQUIRED.
                      (FOLLOWING ALPHA DECAY EVENT).
                   O: THOUGH PRESENT STATUS OF ACCURACY SEEMED TO MEET THE REQUIREMENT CONFIRMATION IS
                      REQUIRED.  ASSAY OF PU ISOTOPES BY GAMMA-RAY SPECTROSCOPY.

31 094 PLUTONIUM 239     NEUTRON                    TOTAL CROSS SECTION
   01  760401    801110    761010
           1.00 KEV   200.00 KEV     2.0%   1    NIG   M.KAWAI
       780198  762210   FISSION REACTOR
                   O: FISSION REACTOR CALCULATIONS.

32 094 PLUTONIUM 239     NEUTRON                    FISSION CROSS SECTION
   01  760401    801110    761010
          10.00 KEV    20.00 MEV     3.0%   1    NIG   M.KAWAI
       780199  762211   FISSION REACTOR
                   O: FISSION REACTOR CORE DESIGN AND ANALYSIS.  LARGE DISCREPANCY BETWEEN
                      EXPERIMENTAL DATA FROM 50 KEV TO 1.0 MEV.

33 094 PLUTONIUM 239     NEUTRON                    NEUTRONS EMITTED PER FISSION(NU BAR)
   01  700401    801110    701010
             UP TO    15.00 MEV     0.5%   1    NIG   M.KAWAI
       780201  702037   FISSION REACTOR
                   A: ACCURAY REQUIRED TO BETTER THAN 0.2 PER CENT, IF POSSIBLE.
                   O: FOR FAST AND HYBRID FUSION REACTOR CALCULATIONS. DISCREPANCY EXISTS BETWEEN
                      EXPERIMENTAL DATA.
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34 094 PLUTONIUM 239                                SPECIAL QUANTITY
   01  760401    801110    761010
                                     0.5%   1    PNC   S.SUZUKI
       780204  762019   SAFEGUARDS
                   Q: DECAY HEAT (W/G) REQUIRED.
                   O: ASSAY OF PU BY CALORIMETRY.

35 094 PLUTONIUM 240                                GAMMA RAY YIELD
   00  760401              761010
                                     1.0%   1    JAE   T.SUZUKI
       780206  762011   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 45.2,104.2 AND 642.3 KEV GAMMA-RAYS REQUIRED.
                      (FOLLOWING ALPHA DECAY EVENT).
                   O: THOUGH PRESENT STATUS OF ACCURACY SEEMED TO MEET THE REQUIREMENT CONFIRMATION IS
                      REQUIRED.  ASSAY OF PU ISOTOPES BY GAMMA-RAY SPECTROSCOPY.

36 094 PLUTONIUM 240     NEUTRON                    FISSION CROSS SECTION
   01  760401    801110    761010
         0.25-1 EV      1.00 MEV    10.0%   1    PNC   M.SASAKI
       780209  762213   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

37 094 PLUTONIUM 240     NEUTRON                    RESONANCE PARAMETERS
   01  760401    801110    761010
           1.00 EV      10.00 KEV            1    PNC   M.SASAKI
       780211  762215   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

38 094 PLUTONIUM 240                                SPECIAL QUANTITY
   01  760401    801110    761010
                                     0.5%   1    PNC   S.SUZUKI
       780212  762020   SAFEGUARDS
                   Q: DECAY HEAT (W/G) REQUIRED.
                   O: ASSAY OF PU BY CALORIMETRY.

39 094 PLUTONIUM 241     NEUTRON                    GAMMA RAY YIELD
   00  760401              761010
                                     5.0%   1    JAE   T.SUZUKI
       780213  762012   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 56.4,77,103.5,148.6 AND 160 KEV GAMMA-RAYS REQUIRED.
                      (FOLLOWING ALPHA DECAY EVENT).
                   A: 1 PER CENT ACCURACY FOR 103.5 AND 148.6 KEV GAMMA RAYS, 5 PER CENT ACCURACY FOR
                      56.4,77 AND 160 KEV GAMMA RAYS.
                   O: THOUGH PRESENT STATUS OF ACCURACY SEEMED TO MEET THE REQUIREMENT CONFIRMATION IS
                      REQUIRED.  ASSAY OF PU ISOTOPES BY GAMMA-RAY SPECTROSCOPY.
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40 094 PLUTONIUM 241                          SPECIAL QUANTITY
   01  760401    801110    761010
                                   0.5%   1    PNC    S.SUZUKI
       780222  762021   SAFEGUARDS
                   Q: DECAY HEAT (W/G) REQUIRED.
                   O: ASSAY OF PU BY CALORIMETRY.

41 094 PLUTONIUM 242                          SPECIAL QUANTITY
   01  760401    801110    761010
                                   0.5%   1    PNC    S.SUZUKI
       780227  762022   SAFEGUARDS
                   Q: DECAY HEAT (W/G) REQUIRED.
                   O: ASSAY OF PU BY CALORIMETRY.

42 095 AMERICIUM 241     NEUTRON              CAPTURE CROSS SECTION
   01  750401    801110    751010
         500.00 KEV    15.00 MEV    10.0%   1    PNC    R.YUMOTO            SAE    H.MATSUNOBU
                                                 MAP    T.HOJUYAMA
       780230  752033   FISSION REACTOR
                   Q: PRODUCTION OF AM-242 AND AM-242M WANTED.
                   O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                      SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.

43 095 AMERICIUM 241                          SPECIAL QUANTITY
   01  760401    801110    761010
                                   0.5%   1    PNC    S.SUZUKI
       780231  762023   SAFEGUARDS
                   Q: DECAY HEAT (W/G) REQUIRED.
                   O: ASSAY OF PU BY CALORIMETRY.

44 095 AMERICIUM 243     NEUTRON              CAPTURE CROSS SECTION
   00  801110              801225
          20.00 EV     15.00 MEV            1    PNC    R.YUMOTO            SAE    H.MATSUNOBU
                                                 JAE    R.SHINDO            MAP    T.HOJUYAMA
       800027           FISSION REACTOR
                   Q: CAPTURE CROSS SECTIONS TO GROUND AND ISOMER STATES OF AM-244 REQUIRED.  THE
                      EXPERIMENTAL DATA ARE VERY SCARCE IN KEV AND MEV REGIONS.
                   O: FOR BURNUP CALCULATIONS OF THERMAL AND FAST REACTORS, ESTIMATION FOR BUILD-UP OF
                      TRANSURANIUM NUCLIDES IN SPENT FUEL, AND NEUTRON SHIELDING DESIGN FOR TRANSPORT
                      CASK OF SPENT FUEL.

45 096 CURIUM 242        NEUTRON              CAPTURE CROSS SECTION
   00  750401              751010
          0.25-1 EV     15.00 MEV            1    PNC    R.YUMOTO            SAE    H.MATSUNOBU
                                                 MAP    T.HOJUYAMA
       780238  752042   FISSION REACTOR
                   A: ACCURACY REQUIRED 10 TO 20 PER CENT.
                   O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                      SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.
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46 096 CURIUM 242        NEUTRON                FISSION CROSS SECTION
   00  750401              751010
         0.25-1 EV     15.00 MEV            1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
                                                  MAP   T.HOJUYAMA
       780240  752041   FISSION REACTOR
                    A: ACCURACY REQUIRED 10 TO 20 PER CENT.
                    O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                       SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.

47 096 CURIUM 243        NEUTRON                CAPTURE CROSS SECTION
   00  750401              751010
          20.00 EV     10.00 MEV            1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
       780241  752047   FISSION REACTOR
                    A: ACCURACY REQUIRED 10 TO 20 PER CENT.
                    O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                       SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.

48 096 CURIUM 243        NEUTRON                FISSION CROSS SECTION
   00  750401              751010
           3.00 MEV     10.00 MEV            1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
       780243  752045   FISSION REACTOR
                    A: ACCURACY REQUIRED 10 TO 20 PER CENT.
                    O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                       SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.
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4. List of Withdrawn Requests from WRENDA 79/80

Withdrawn requests of 111 are shown in this section. They were carefully reviewed by the requestors before they were decided to be withdrawn. Reasons for withdrawal are satisfaction of the requirements, changes of the requestors' plan in which they were going to use the data, amalgamation of some requests which were made for similar quantities, and so forth.

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                    ** REQUESTS WITHDRAWN FROM WRENDA.        **                          PAGE    1

1 003 LITHIUM 006       NEUTRON                ELASTIC CROSS SECTION
  00  760401              761010    801110
          7.50 MEV    15.00 MEV    10.0%   2    JAE   Y.SEKI
      780001  762168   FUSION REACTOR
                  0: NEUTRON TRANSPORT CALCULATIONS.

2 003 LITHIUM 006       NEUTRON                DIFFERENTIAL ELASTIC CROSS SECTION
  00  760401              761010    801110
          7.50 MEV    15.00 MEV    10.0%   2    JAE   Y.SEKI
      780002  762051   FUSION REACTOR
                  0: NEUTRON TRANSPORT CALCULATIONS.

3 003 LITHIUM 007       NEUTRON                ELASTIC CROSS SECTION
  00  760401              761010    801110
          7.50 MEV    15.00 MEV     5.0%   2    JAE   Y.SEKI
      780006  762230   FUSION REACTOR
                  0: NEUTRON TRANSPORT CALCULATIONS.

4 003 LITHIUM 007       NEUTRON                DIFFERENTIAL ELASTIC CROSS SECTION
  00  760401              761010    801110
          7.50 MEV    15.00 MEV    10.0%   2    JAE   Y.SEKI
      780007  762055   FUSION REACTOR
                  0: NEUTRON TRANSPORT CALCULATIONS.

5 003 LITHIUM 007       NEUTRON                INELASTIC CROSS SECTION
  00  760401              761010    801110
            UP TO     15.00 MEV    15.0%   2    JAE   Y.SEKI
      780008  762231   FUSION REACTOR
                  0: NEUTRON TRANSPORT CALCULATIONS.

6 003 LITHIUM 007       NEUTRON                ENERGY DIFFERENTIAL INELASTIC CROSS SECTION
  00  760401              761010    801110
            UP TO     15.00 MEV    15.0%   2    JAE   Y.SEKI
      780009  762056   FUSION REACTOR
                  0: NEUTRON TRANSPORT CALCULATIONS.

7 003 LITHIUM 007       NEUTRON                N,2N ANGULAR DISTRIBUTION
  00  760401              761010    801110
            UP TO     15.00 MEV    15.0%   2    JAE   Y.SEKI
      780011  762232   FUSION REACTOR
                  0: BLANKET NEUTRONICS CALCULATIONS.

8 003 LITHIUM 007       NEUTRON                N,2N NEUTRON SPECTRA
  00  760401              761010    801110
            UP TO     15.00 MEV    15.0%   2    JAE   Y.SEKI
      780012  762057   FUSION REACTOR
                  0: BLANKET NEUTRONICS CALCULATIONS.
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                    ** REQUESTS WITHDRAWN FROM WRENDA.        **                          PAGE    2

 9 004 BERYLLIUM 009     NEUTRON                INELASTIC CROSS SECTION
   00  760401              761010    801110
             UP TO    15.00 MEV    15.0%   2    JAE   Y.SEKI
       780014  762060   FUSION REACTOR
                   O: BLANKET NEUTRONICS CALCULATIONS.

10 004 BERYLLIUM 009     NEUTRON                N,2N
   00  760401              761010    801110
             UP TO    15.00 MEV    15.0%   3    JAE   Y.SEKI              MAP   M.KASAI
       780015  762061   FUSION REACTOR
                   O: NEUTRON MULTIPLICATION CALCULATIONS.

11 004 BERYLLIUM 009     NEUTRON                N,2N ANGULAR DISTRIBUTION
   00  760401              761010    801110
             UP TO    15.00 MEV    15.0%   2    JAE   Y.SEKI
       780016  762233   FUSION REACTOR
                   O: NEUTRON TRANSPORT CALCULATIONS.

12 004 BERYLLIUM 009     NEUTRON                N,2N NEUTRON SPECTRA
   00  760401              761010    801110
             UP TO    15.00 MEV    15.0%   3    JAE   Y.SEKI
       780017  762062   FUSION REACTOR
                   O: NEUTRON TRANSPORT CALCULATIONS.

13 006 CARBON 012        NEUTRON                INELASTIC CROSS SECTION
   00  760401              761010    801110
           8.00 MEV    15.00 MEV    10.0%   2    JAE   Y.SEKI
       780019  762064   FUSION REACTOR
                   Q: INELASTICALLY SCATTERED NEUTRON SPECTRA REQUIRED WITH INCIDENT ENERGY STEPS 0.5
                      MEV.
                   O: NEUTRON TRANSPORT CALCULATIONS.

14 013 ALUMINIUM 027     NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    15.0%   3    MAP   M.KASAI
       780031  762074   FUSION REACTOR
                   O: GAMMA-RAY HEATING CALCULATIONS.

15 013 ALUMINIUM 027     NEUTRON                N,2N
   00  760401              761010    801110
             UP TO    15.00 MEV    15.0%   3    MAP   M.KASAI
       780033  762070   FUSION REACTOR
                   O: POTENTIAL CONSTITUENT FOR STRUCTURAL MATERIAL.  NEUTRON MULTIPLICATION
                      CALCULATIONS.
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16 013 ALUMINIUM 027      NEUTRON                N,P
   00  760401               761010    801110
             UP TO     15.00 MEV    15.0%   3    MAP   M.KASAI
       780034  762071    FUSION REACTOR
                    O: HYDROGEN ACCUMULATION CALCULATIONS.

17 018 ARGON 036          NEUTRON                N,P
   00  760401               761010    801110
         0.25-1 EV      15.00 MEV    30.0%   2    MAP   T.NISIMURA
       780038  762177    FISSION REACTOR
                    O: FOR FBR SHIELDING CALCULATIONS.  FOR FBR SAFETY ANALYSIS.

18 020 CALCIUM            NEUTRON                CAPTURE CROSS SECTION
   00  760401               761010    801110
         0.25-1 EV      15.00 MEV    15.0%   3    JAE   Y.SEKI
       780043  762077    FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: INCLUDED IN CONCRETE.  SHIELDING DESIGN AND GAMMA-RAY HEATING CALCULATION.

19 022 TITANIUM           NEUTRON                INELASTIC CROSS SECTION
   00  760401               761010    801110
             UP TO     15.00 MEV    15.0%   3    MAP   M.KASAI
       780045  762079    FUSION REACTOR
                    O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  NEUTRON TRANSPORT CALCULATIONS.

20 022 TITANIUM           NEUTRON                N,2N
   00  760401               761010    801110
             UP TO     15.00 MEV    15.0%   3    MAP   M.KASAI
       780047  762080    FUSION REACTOR
                    O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  NEUTRON  MULTIPLICATION
                       CALCULATIONS.

21 022 TITANIUM           NEUTRON                N,P
   00  760401               761010    801110
             UP TO     15.00 MEV    15.0%   3    MAP   M.KASAI
       780048  762081    FUSION REACTOR
                    O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  HYDROGEN ACCUMULATION
                       CALCULATIONS.

22 023 VANADIUM           NEUTRON                INELASTIC CROSS SECTION
   00  760401               761010    801110
             UP TO     15.00 MEV    10.0%   2    MAP   M.KASAI
       780050  762084    FUSION REACTOR
                    O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  NEUTRON TRANSPORT CALCULATIONS.
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23 023 VANADIUM          NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    10.0%   2    MAP   K.IOKI
       780051  762088   FUSION REACTOR
                  O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  GAMMA-RAY HEATING CALCULATIONS.

24 023 VANADIUM          NEUTRON                N,P
   00  760401              761010    801110
             UP TO     15.00 MEV    10.0%   2    MAP   M.KASAI              MAP   K.IOKI
       780054  762086   FUSION REACTOR
                  O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  HYDROGEN ACCUMULATION
                     CALCULATIONS.

25 023 VANADIUM          NEUTRON                N,ALPHA
   00  760401              761010    801110
             UP TO     15.00 MEV    10.0%   2    MAP   M.KASAI              MAP   K.IOKI
       780055  762087   FUSION REACTOR
                  O: POTENTIAL CONSTITUENT OF STRUCTURAL MATERIAL.  HELIUM ACCUMULATION AND NEUTRON
                     TRANSPORT CALCULATIONS.

26 023 VANADIUM 050      NEUTRON                N,2N
   00  760401              761010    801110
             UP TO     15.00 MEV    10.0%   3    MAP   M.KASAI
       780056  762091   FUSION REACTOR
                  O: TRANSMUTATION CALCULATIONS.

27 023 VANADIUM 050      NEUTRON                N,ALPHA
   00  760401              761010    801110
             UP TO     15.00 MEV    10.0%   3    MAP   K.IOKI               MAP   M.KASAI
       780057  762092   FUSION REACTOR
                  O: TRANSMUTATION CALCULATIONS.

28 024 CHROMIUM          NEUTRON                INELASTIC CROSS SECTION
   00  760401              761010    801110
             UP TO     15.00 MEV    15.0%   2    JAE   Y.SEKI
       780058  762093   FUSION REACTOR
                  Q: INELASTIC GAMMA-RAY SPECTRA ALSO REQUIRED.
                  O: NEUTRON TRANSPORT AND GAMMA-RAY HEATING CALCULATIONS.

29 024 CHROMIUM 052      NEUTRON                N,2N
   00  760401              761010    801110
             UP TO     15.00 MEV    15.0%   3    MAP   M.KASAI
       780063  762098   FUSION REACTOR
                  O: TRANSMUTATION CALCULATIONS.
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30 026 IRON                NEUTRON                CAPTURE CROSS SECTION
   00  760401                761010    801110
         0.25-1 EV      15.00 MEV    15.0%   2    JAE   Y.SEKI
       780065  762100   FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: NEUTRON TRANSPORT AND GAMMA-RAY HEATING CALCULATIONS.

31 026 IRON 058            NEUTRON                CAPTURE CROSS SECTION
   00  760401                761010    801110
         0.25-1 EV      15.00 MEV    20.0%   2    NIG   M.KAWAI
       780070  762179   FISSION REACTOR
                    O: FOR RADIATION SHIELDING TO 1.2916 MEV GAMMA-RAY FROM IRON-59 IN CORROSION
                       PRODUCTS.

32 027 COBALT 059          NEUTRON                CAPTURE CROSS SECTION
   00  710501                711010    801110
             UP TO      10.00 MEV            2    NIG   M.KAWAI
       780071  712028   FISSION REACTOR
                    A: ACCURACY REQUIRED TO BETTER THAN 20.0 PERCENT.
                    O: FOR FUEL CASK DESIGN AND CONTROL ROD DESIGN.

33 028 NICKEL              NEUTRON                CAPTURE CROSS SECTION
   00  760401                761010    801110
         0.25-1 EV      15.00 MEV    15.0%   2    JAE   Y.SEKI
       780073  762110   FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: GAMMA-RAY HEATING CALCULATIONS.

34 028 NICKEL              NEUTRON                N,T
   00  760401                761010    801110
             UP TO      15.00 MEV    15.0%   3    MAP   M.KASAI
       780077  762109   FUSION REACTOR
                    O: TRANSMUTATION CALCULATIONS.

35 029 COPPER              NEUTRON                CAPTURE CROSS SECTION
   00  760401                761010    801110
         0.25-1 EV      15.00 MEV    15.0%   2    JAE   Y.SEKI
       780079  762114   FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: GAMMA-RAY HEATING IN MAGNETS.

36 029 COPPER              NEUTRON                PHOTON PRODUCTION CROSS SECTION IN INELASTIC SCAT
   00  760401                761010    801110
             UP TO      15.00 MEV    15.0%   2    JAE   Y.SEKI
       780080  762112   FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: GAMMA-RAY HEATING IN MAGNETS.
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37 041 NIOBIUM 092        NEUTRON                N,ALPHA
   00  760401               761010    801110
             UP TO     15.00 MEV    30.0%   3    MAP    K.IOKI
       780088  762115    FUSION REACTOR
                    O: TRANSMUTATION CALCULATIONS.

38 041 NIOBIUM 093        NEUTRON                CAPTURE CROSS SECTION
   00  760401               761010    801110
         0.25-1 EV     15.00 MEV    15.0%   2    JAE    Y.SEKI
       780090  762122    FUSION REACTOR
                    Q: GAMMA-RAY SPECTRA ALSO REQUIRED.
                    O: GAMMA-RAY HEATING CALCULATIONS.

39 041 NIOBIUM 093        NEUTRON                CAPTURE CROSS SECTION
   00  760401               761010    801110
         0.25-1 EV     15.00 MEV    20.0%   3    MAP    M.KASAI
       780091  762123    FUSION REACTOR
                    Q: CAPTURE CROSS SECTION TO NB-94M IS REQUIRED.
                    O: TRANSMUTATION CALCULATIONS.

40 041 NIOBIUM 093        NEUTRON                N,2N
   00  760401               761010    801110
             UP TO     15.00 MEV    10.0%   2    MAP    M.KASAI
       780093  762118    FUSION REACTOR
                    O: NEUTRON MULTIPLICATION CALCULATIONS.

41 041 NIOBIUM 093        NEUTRON                N,ALPHA
   00  760401               761010    801110
             UP TO     15.00 MEV    15.0%   2    MAP    M.KASAI              MAP    K.IOKI
       780095  762120    FUSION REACTOR
                    O: HELIUM ACCUMULATION CALCULATIONS.

42 041 NIOBIUM 094        NEUTRON                CAPTURE CROSS SECTION
   00  760401               761010    801110
         0.25-1 EV     15.00 MEV    10.0%   3    MAP    M.KASAI
       780097  762125    FUSION REACTOR
                    O: TRANSMUTATION CALCULATIONS.

43 042 MOLYBDENUM         NEUTRON                ELASTIC CROSS SECTION
   00  760401               761010    801110
            1.00 MEV     15.00 MEV    10.0%   2    JAE    Y.SEKI
       780098  762235    FUSION REACTOR
                    Q: CROSS SECTIONS FOR EACH ISOTOPE ARE REQUESTED.
                    O: NEUTRON TRANSPORT CALCULATIONS.
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                    ** REQUESTS WITHDRAWN FROM WRENDA.        **                          PAGE    7

44 042 MOLYBDENUM        NEUTRON                INELASTIC CROSS SECTION
   00  760401              761010    801110
             UP TO    15.00 MEV    15.0%   2    JAE   Y.SEKI
       780100  762236   FUSION REACTOR
                   Q: CROSS SECTIONS FOR EACH ISOTOPE ARE REQUESTED.  GAMMA-RAY SPECTRA ALSO REQUIRED.
                   O: NEUTRON TRANSPORT CALCULATIONS.

45 042 MOLYBDENUM        NEUTRON                ENERGY DIFFERENTIAL INELASTIC CROSS SECTION
   00  760401              761010    801110
             UP TO    15.00 MEV    15.0%   2    JAE   Y.SEKI
       780101  762127   FUSION REACTOR
                   Q: CROSS SECTIONS FOR EACH ISOTOPE ARE ALSO REQUESTED. GAMMA-RAY SPECTRA ALSO
                      REQUIRED.
                   O: NEUTRON TRANSPORT CALCULATIONS.

46 042 MOLYBDENUM        NEUTRON                N,2N
   00  760401              761010    801110
             UP TO    15.00 MEV    10.0%   2    JAE   Y.SEKI
       780103  762128   FUSION REACTOR
                   Q: CROSS SECTIONS FOR EACH ISOTOPE ARE ALSO REQUESTED.
                   O: NEUTRON TRANSPORT CALCULATIONS.

47 042 MOLYBDENUM 092    NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    10.0%   2    MAP   K.IOKI
       780106  762132   FUSION REACTOR
                   O: NEUTRON BALANCE AND TRANSMUTATION CALCULATIONS.

48 042 MOLYBDENUM 092    NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    20.0%   2    MAP   T.HOJUYAMA
       780107  762181   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

49 042 MOLYBDENUM 094    NEUTRON                TOTAL CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    10.0%   2    MAP   T.HOJUYAMA
       780109  762183   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

50 042 MOLYBDENUM 094    NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV    100.00 KEV    20.0%   2    MAP   T.HOJUYAMA
       780110  762184   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.
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51 042 MOLYBDENUM 094    NEUTRON                N,P
   00  760401              761010    801110
             UP TO    15.00 MEV    30.0%   2    MAP   T.HOJUYAMA
       780112  762186   FISSION REACTOR
                   0: FOR FAST REACTOR CALCULATIONS.

52 042 MOLYBDENUM 094    NEUTRON                N,ALPHA
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    30.0%   2    MAP   T.HOJUYAMA
       780113  762187   FISSION REACTOR
                   0: FOR FAST REACTOR CALCULATIONS.

53 042 MOLYBDENUM 095    NEUTRON                TOTAL CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    10.0%   2    MAP   T.HOJUYAMA
       780114  762188   FISSION REACTOR
                   0: FOR FAST REACTOR CALCULATIONS.

54 042 MOLYBDENUM 095    NEUTRON                INELASTIC CROSS SECTION
   00  760401              761010    801110
             UP TO    15.00 MEV    20.0%   2    MAP   T.HOJUYAMA
       780115  762189   FISSION REACTOR
                   0: FOR FAST REACTOR CALCULATIONS.

55 042 MOLYBDENUM 095    NEUTRON                N,ALPHA
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    20.0%   2    MAP   T.HOJUYAMA
       780116  762191   FISSION REACTOR
                   0: FOR FAST REACTOR CALCULATIONS.

56 042 MOLYBDENUM 096    NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    20.0%   2    MAP   T.HOJUYAMA
       780117  762193   FISSION REACTOR
                   0: FOR FAST REACTOR CALCULATIONS.

57 042 MOLYBDENUM 096    NEUTRON                N,ALPHA
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    30.0%   2    MAP   T.HOJUYAMA
       780118  762195   FISSION REACTOR
                   0: FOR FAST REACTOR CALCULATIONS.

58 042 MOLYBDENUM 097    NEUTRON                INELASTIC CROSS SECTION
   00  760401              761010    801110
             UP TO    15.00 MEV    30.0%   2    MAP   T.HOJUYAMA
       780119  762197   FISSION REACTOR
                   0: FOR FAST REACTOR CALCULATIONS.
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59 042 MOLYBDENUM 097    NEUTRON                N,ALPHA
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    30.0%   2    MAP   T.HOJUYAMA
       780120  762198   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

60 042 MOLYBDENUM 097    NEUTRON                TOTAL CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    10.0%   2    MAP   T.HOJUYAMA
       781118  762196   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

61 042 MOLYBDENUM 098    NEUTRON                N,ALPHA
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    30.0%   2    MAP   T.HOJUYAMA
       780121  762200   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

62 042 MOLYBDENUM 100    NEUTRON                N,P
   00  760401              761010    801110
             UP TO     15.00 MEV    30.0%   2    MAP   T.HOJUYAMA
       780122  762203   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

63 042 MOLYBDENUM 100    NEUTRON                N,ALPHA
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    30.0%   2    MAP   T.HOJUYAMA
       780123  762204   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.

64 044 RUTHENIUM 101     NEUTRON                CAPTURE CROSS SECTION
   00  750401              751010    801110
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780125  752008   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.  EVALUATIONS ARE VERY DISCREPANT.

65 045 RHODIUM 106                              GAMMA RAY YIELD
   00  720401              721010    801110
                                     1.0%   2    JAE   K.TASAKA
       780128  722004   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 512, 616, 622 AND 1050 KEV GAMMA-RAYS REQUIRED.
                      (FOLLOWING BETA DECAY EVENT).
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.
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66 046 PALLADIUM 105     NEUTRON                CAPTURE CROSS SECTION
   00  750401              751010    801110
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780129  752011   FISSION REACTOR
                   Q: EVALUATION ALSO REQUIRED.
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.  DATA BETWEEN 160 EV TO A FEW KEV ARE
                      LACKING.  NO EXPERIMENTAL DATA ABOVE 100 EV.

67 047 SILVER 109        NEUTRON                CAPTURE CROSS SECTION
   00  750401              751010    801110
         100.00 EV    500.00 KEV    10.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780131  752013   FISSION REACTOR
                   A: 10.0 TO 20.0 %.
                   O: FOR FAST REACTOR CALCULATIONS.  DISCREPANT SERIES OF DATA IN KEV REGION.  DATA
                      ARE ALSO AVAILABLE FOR AG-107 AND NATURAL AG.

68 051 ANTIMONY 121      NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    15.0%   2    MAP   T.HOJUYAMA
       780133  762205   FISSION REACTOR
                   O: FOR NEUTRON SOURCE CALCULATION.

69 051 ANTIMONY 123      NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
         0.25-1 EV     15.00 MEV    15.0%   2    MAP   T.HOJUYAMA
       780134  762206   FISSION REACTOR
                   O: FOR NEUTRON SOURCE CALCULATION.

70 051 ANTIMONY 125                             GAMMA RAY YIELD
   00  720401              721010    801110
                                     1.0%   2    JAE   K.TASAKA
       780136  722006   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 176,381,428,464,601,607,636, AND 672 KEV GAMMA-RAYS
                      REQUIRED.  (FOLLOWING BETA DECAY EVENT).
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

71 054 XENON 131         NEUTRON                RESONANCE PARAMETERS
   00  780808              781212    801110
         100.00 EV    500.00 KEV    20.0%   2    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780143  792069   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.  NO KEV DATA AT ALL .  EVALUATIONS ARE
                      VERY DISCREPANT.  EVALUATIONS ARE ALSO REQUIRED.

72 055 CESIUM 133        NEUTRON                CAPTURE CROSS SECTION
   00  720401              721010    801110
         0.25-1 EV     14.00 MEV     3.0%   1    JAE   H.OKASHITA
       780145  722021   SAFEGUARDS
                   Q: RESONANCE INTEGRAL ALSO WANTED.
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.
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73 055 CESIUM 133        NEUTRON                CAPTURE CROSS SECTION
   00  750401              751010    801110
         100.00 EV    500.00 KEV    10.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780146  752015   FISSION REACTOR
                   O: FOR FAST REACTOR BURNUP CALCULATIONS.  EVALUATIONS ARE ALSO REQUIRED.
                      SYSTEMATIC DISCREPANCY BETWEEN THE KEV DATA SETS.

74 055 CESIUM 134                               GAMMA RAY YIELD
   00  720401              721010    801110
                                     1.0%   2    JAE   H.OKASHITA
       780147  722007   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 563,569,796,802 AND 1365 KEV GAMMA-RAYS REQUIRED.
                      (FOLLOWING BETA DECAY EVENT).
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

75 056 BARIUM 133                               SPECIAL QUANTITY
   00  760401              761010    801110
                                     3.0%   3    TIT   K.HISATAKE
       780151  762207   FISSION REACTOR
                   Q: RELATIVE YIELDS OF 53.2,79.6,81.0,160.6,276.4,302.0 AND 356.0 KEV GAMMA-RAYS.
                   O: INTENSITY STANDARDS FOR GAMMA-RAY MEASUREMENTS.

76 057 LANTHANUM 140                            GAMMA RAY YIELD
   00  720401              721010    801110
                                     1.0%   2    JAE   K.TASAKA
       780152  722009   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 328.8,487.0,815.8 AND 2522.0 KEV GAMMA-RAYS
                      REQUIRED.  (FOLLOWING BETA DECAY EVENT).
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

77 058 CERIUM 144                               GAMMA RAY YIELD
   00  720401              721010    801110
                                     1.0%   2    JAE   H.OKASHITA
       780153  722011   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 133.5 KEV GAMMA-RAY REQUIRED. (FOLLOWING BETA DECAY
                      EVENT).
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

78 059 PRASEODYMIUM 141  NEUTRON                CAPTURE CROSS SECTION
   00  720401              721010    801110
         0.25-1 EV     14.00 MEV     3.0%   1    JAE   H.OKASHITA
       780154  722023   SAFEGUARDS
                   Q: RESONANCE INTEGRAL ALSO WANTED.
                   O: FOR BURNUP CALCULATION FROM DESTRUCTIVE MEASUREMENT.
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79 059 PRASEODYMIUM 144                          GAMMA RAY YIELD
   00  720401              721010    801110
                                     1.0%   1    JAE   H.OKASHITA
       780155  722012   SAFEGUARDS
                   Q: YIELD PER DISINTEGRATION OF 696.5,1498.1 AND 2185.7 KEV GAMMA-RAYS REQUIRED.
                      (FOLLOWING BETA DECAY EVENT).
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

80 060 NEODYMIUM 143     NEUTRON                 CAPTURE CROSS SECTION
   00  750401              751010    801110
         100.00 EV    400.00 KEV    20.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780156  752017   FISSION REACTOR
                   Q: DESIRED WITH LOWER PRIORITY FOR WIDER ENERGY RANGE.
                   O: FOR FAST REACTOR CALCULATIONS.

81 060 NEODYMIUM 145     NEUTRON                 CAPTURE CROSS SECTION
   00  750401              751010    801110
         100.00 EV    400.00 KEV    20.0%   1    NIG   S.IIJIMA            SAE   H.MATSUNOBU
       780157  752018   FISSION REACTOR
                   Q: DESIRED WITH LOWER PRIORITY FOR WIDER ENERGY RANGE.
                   O: FOR FAST REACTOR CALCULATIONS.  NO EXPERIMENTAL DATA FROM 100 EV TO 400 KEV.

82 063 EUROPIUM 153      NEUTRON                 CAPTURE CROSS SECTION
   00  720401              721010    801110
         0.25-1 EV     14.00 MEV     5.0%   1    JAE   H.OKASHITA
       780161  722038   SAFEGUARDS
                   Q: RESONANCE INTEGRAL ALSO WANTED.
                   O: FOR BURNUP CALCULATION FROM NON-DESTRUCTIVE MEASUREMENT.

83 092 URANIUM 235       GAMMA                   FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401              761010    801110
           4.00 MEV    14.00 MEV    10.0%   3    KKU   R.MIKI
       780171  762034   SAFEGUARDS
                   Q: TOTAL FISSION YIELD PRODUCED BY BREMSSTRAHLUNG REQUIRED. YIELD MAY BE IN THE
                      UNIT OF YIELD ROENTGEN*NUCLEUS OR RELATIVE TO U-238 OR OTHER PHOTO ACTIVATION
                      YIELDS.
                   O: BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) OF SUFFICIENT THICKNESS TO STOP
                      ELECTRONS.  NON-DESTRUCTIVE ASSAY OF U.

84 092 URANIUM 235       GAMMA                   FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401              761010    801110
           4.00 MEV    14.00 MEV     5.0%   3    KKU   R.MIKI
       780172  762042   SAFEGUARDS
                   Q: CUMULATIVE YIELDS OF HIGH FISSION YIELD ISOTOPES.
                   O: BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) OF SUFFICIENT THICKNESS TO STOP
                      ELECTRONS.  NON-DESTRUCTIVE ASSAY OF NUCLEAR MATERIALS.
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85 092 URANIUM 238        GAMMA                    FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401               761010     801110
          4.00 MEV     14.00 MEV     10.0%   3     KKU    R.MIKI
       780177  762035    SAFEGUARDS
                    Q: TOTAL FISSION YIELD PRODUCED BY BREMSSTRAHLUNG REQUIRED. YIELD MAY BE IN THE
                       UNIT OF YIELD/ROENTGEN*NUCLEUS OR RELATIVE TO OTHER PHOTOACTIVATION YIELDS.
                    O: BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) SUFFICIENT THICKNESS TO STOP ELECTRONS.
                       NON-DESTRUCTIVE ASSAY OF U.

86 092 URANIUM 238        GAMMA                    FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401               761010     801110
          4.00 MEV     14.00 MEV      5.0%   3     KKU    R.MIKI
       780178  762043    SAFEGUARDS
                    Q: CUMULATIVE YIELDS OF HIGH FISSION YIELD ISOTOPES.
                    O: NON-DESTRUCTIVE ASSAY OF NUCLEAR MATERIALS.

87 093 NEPTUNIUM 239      NEUTRON                  CAPTURE CROSS SECTION
   00  760401               761010     801110
        0.25-1 EV      10.00 MEV     10.0%   3     NFI    M.YADA
       780185  762025    SAFEGUARDS
                    O: FOR HIGHER BURNUP CALCULATIONS.

88 093 NEPTUNIUM 239      NEUTRON                  FISSION CROSS SECTION
   00  760401               761010     801110
        0.25-1 EV      10.00 MEV     25.0%   3     NFI    M.YADA
       780187  762032    SAFEGUARDS
                    Q: THE VALUE OF NU ALSO WANTED.
                    A: 10 PER CENT ACCURACY IS DESIRABLE FOR APPLICATION.
                    O: NO EXPERIMENTAL DATA.  BURNUP ANALYSIS OF FAST BREEDER REACTORS.

89 094 PLUTONIUM 238      GAMMA                    FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401               761010     801110
          4.00 MEV     14.00 MEV     10.0%   3     KKU    R.MIKI
       780190  762036    SAFEGUARDS
                    Q: TOTAL FISSION YIELD PRODUCED BY BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) OF
                       SUFFICIENT THICKNESS TO STOP ELECTRONS .  NO EXPERIMENTAL DATA.
                    O: NON-DESTRUCTIVE ASSAY OF U.

90 094 PLUTONIUM 239      SPONTANEOUS              FISSION HALF LIFE
   01  760401    801110     761010     801110
                                        1.0%   2     PNC    S.SUZUKI
       780194  762015    SAFEGUARDS
                    O: DETECTION OF PU BY NEUTRON COINCIDENCE METHOD.
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91 094 PLUTONIUM 239     GAMMA                  FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401              761010    801110
          4.00 MEV    14.00 MEV    10.0%   3    KKU   R.MIKI
       780196  762037   SAFEGUARDS
                   Q: TOTAL FISSION YIELD PRODUCED BY BREMSSTRAHLUNG REQUIRED. YIELD MAY BE IN THE
                      UNIT OF YIELD/ROENTGEN*NUCLEUS OR RELATIVE TO U-238 OR OTHER PHOTOACTIVATION
                      YIELDS.
                   O: BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) OF SUFFICIENT THICKNESS TO STOP
                      ELECTRONS.  NON-DESTRUCTIVE ASSAY OF PU.

92 094 PLUTONIUM 239     GAMMA                  FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401              761010    801110
          4.00 MEV    14.00 MEV     5.0%   3    KKU   R.MIKI
       780197  762045   SAFEGUARDS
                   Q: CUMULATIVE YIELDS OF HIGH FISSION YIELD ISOTOPES.
                   O: BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) OF SUFFICIENT THICKNESS TO STOP
                      ELECTRONS.  NON-DESTRUCTIVE ASSAY OF NUCLEAR MATERIALS.

93 094 PLUTONIUM 240     GAMMA                  FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401              761010    801110
          4.00 MEV    14.00 MEV    10.0%   3    KKU   R.MIKI
       780207  762038   SAFEGUARDS
                   Q: TOTAL FISSION YIELD PRODUCED BY BREMSSTRAHLUNG REQUIRED. YIELD MAY BE IN THE
                      UNIT OF YIELD/ROENTGEN*NUCLEUS OR RELATIVE TO U-238 OR OTHER PHOTOACTIVATION
                      YIELDS.
                   O: BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) OF SUFFICIENT THICKNESS TO STOP
                      ELECTRONS.  NO EXPERIMENTAL DATA. NON-DESTRUCTIVE ASSAY OF PU.

94 094 PLUTONIUM 240     NEUTRON                CAPTURE CROSS SECTION
   00  760401              761010    801110
          1.00 KEV   500.00 KEV    10.0%   1    MAP   Y.SEKI
       780208  762214   FISSION REACTOR
                   O: FOR EVALUATION OF BREEDING RATIO AND BURNUP REACTIVITY CHANGE IN FAST REACTOR
                      CALCULATIONS.

95 094 PLUTONIUM 241     GAMMA                  FISSION PRODUCT MASS YIELD SPECTRUM
   00  760401              761010    801110
          4.00 MEV    14.00 MEV    10.0%   3    KKU   R.MIKI
       780214  762039   SAFEGUARDS
                   Q: TOTAL FISSION YIELD PRODUCED BY BREMSSTRAHLUNG REQUIRED. YIELD MAY BE IN THE
                      UNIT OF YIELD/ROENTGEN*NUCLEUS OR RELATIVE TO U-238 OR OTHER PHOTOACTIVATION
                      YIELDS.
                   O: BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) OF SUFFICIENT THICKNESS TO STOP
                      ELECTRONS.  NON-DESTRUCTIVE ASSAY OF PU.

96 094 PLUTONIUM 241     NEUTRON                TOTAL CROSS SECTION
   00  760401              761010    801110
        100.00 EV     15.00 MEV    10.0%   1    MAP   T.HOJUYAMA
       780215  762216   FISSION REACTOR
                   O: FOR FAST REACTOR CALCULATIONS.
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 97 094 PLUTONIUM 241     NEUTRON                CAPTURE CROSS SECTION
    00  760401              761010    801110
            0.10 EV     15.00 MEV     8.0%   1    MAP   T.HOJUYAMA
        780216  762217   FISSION REACTOR
                    O: FOR FAST REACTOR CALCULATIONS.

 98 094 PLUTONIUM 241     NEUTRON                N,2N
    00  760401              761010    801110
                UP TO    15.00 MEV    20.0%   2    MAP   T.HOJUYAMA
        780217  762221   FISSION REACTOR
                    O: FOR FAST REACTOR CALCULATIONS.

 99 094 PLUTONIUM 241     NEUTRON                CAPTURE TO FISSION RATIO(ALPHA)
    00  760401              761010    801110
            0.10 EV     15.00 MEV     8.0%   1    MAP   T.HOJUYAMA
        780219  762219   FISSION REACTOR
                    O: FOR FAST REACTOR CALCULATIONS.

100 094 PLUTONIUM 241     NEUTRON                RESONANCE PARAMETERS
    00  760401              761010    801110
            0.20 EV    200.00 EV     10.0%   1    MAP   T.HOJUYAMA
        780221  762222   FISSION REACTOR
                    A: 10 PER CENT IN FISSION WIDTH.
                    O: FOR FAST REACTOR CALCULATIONS.

101 094 PLUTONIUM 242     NEUTRON                CAPTURE CROSS SECTION
    00  760401              761010    801110
            1.00 KEV    15.00 MEV    10.0%   2    MAP   T.HOJUYAMA
        780225  762223   FISSION REACTOR
                    O: FOR SHIELDING OF SPENT FUEL.

102 094 PLUTONIUM 242     NEUTRON                FISSION CROSS SECTION
    00  760401              761010    801110
            1.00 KEV    15.00 MEV    10.0%   2    MAP   T.HOJUYAMA
        780226  762224   FISSION REACTOR
                    O: FOR SHIELDING OF SPENT FUEL.

103 095 AMERICIUM 241     GAMMA                  FISSION PRODUCT MASS YIELD SPECTRUM
    00  760401              761010    801110
            4.00 MEV    14.00 MEV    10.0%   3    KKU   R.MIKI
        780228  762040   SAFEGUARDS
                    Q: TOTAL FISSION YIELD PRODUCED BY BREMSSTRAHLUNG REQUIRED. YIELD MAY BE IN THE
                       UNIT OF YIELD/ROENTGEN*NUCLEUS OR RELATIVE TO U-238 OR OTHER PHOTOACTIVATION
                       YIELDS.
                    O: BREMSSTRAHLUNG CONVERTER (PREFERABLY TA) OF SUFFICIENT THICKNESS TO STOP
                       ELECTRONS.  NON-DESTRUCTIVE ASSAY OF PU.
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104 095 AMERICIUM 241     NEUTRON                CAPTURE CROSS SECTION
    00  750401              751010    801110
          1.00-2 EV     20.00 EV             1    PNC   R.YUMOTO            SAE   H.MATSUNOBU
                                                   MAP   T.HOJUYAMA
        780229  752032   FISSION REACTOR
                     Q: ENERGY DEPENDENCE WANTED.
                     A: ACCURACY REQUIRED 5 TO 10 PERCENT.
                     O: REACTOR BURNUP CALCULATIONS AND ESTIMATION OF TRANS- URANIUM NUCLIDE BUILD-UP IN
                        SPENT FUEL.  NEUTRON SHIELDING OF SPENT-FUEL TRANSPORT CASK.

105 095 AMERICIUM 242     NEUTRON                CAPTURE CROSS SECTION
    00  720401              761010    801110
          0.25-1 EV     14.00 MEV            2    JAE   Y.NAITO
        780232  722045   SAFEGUARDS
                     A: ACCURACY REQUIRED AT THERMAL IS 10 PER CENT, 20 PER CENT ABOVE.
                     O: FOR BURNUP CALCULATION OF A PU LOADED THERMAL REACTOR.

106 095 AMERICIUM 242     NEUTRON                CAPTURE CROSS SECTION
    00  760401              761010    801110
          0.25-1 EV     10.00 MEV    10.0%   3    NFI   M.YADA
        780233  762026   SAFEGUARDS
                     O: NO MEASUREMENTS OF CAPTURE CROSS SECTION BUT A FEW DATA OF FISSION CROSS SECTION
                        ARE AVAILABLE.  FOR HIGHER BURN- UP CALCULATIONS.

107 095 AMERICIUM 242     NEUTRON                FISSION CROSS SECTION
    00  760401              761010    801110
          0.25-1 EV     10.00 MEV     5.0%   3    NFI   M.YADA
        780235  762033   SAFEGUARDS
                     Q: THE VALUE OF NU ALSO WANTED.
                     A: 10 PER CENT ACCURACY IS DESIRABLE FOR APPLICATION.
                     O: NO EXPERIMENTAL DATA.  THE VALUES OF FISSION CROSS SECTION AND NU ARE KNOWN
                        WITHIN AN ERROR OF 5 PER CENT AT 25.3 MV.  BURNUP ANALYSIS OF FAST BREEDER
                        REACTORS.

108 095 AMERICIUM 243     NEUTRON                FISSION CROSS SECTION
    00  760401              761010    801110
          0.25-1 EV      4.00 MEV    20.0%   1    MAP   T.HOJUYAMA
        780237  762227   FISSION REACTOR
                     O: FOR FAST REACTOR CALCULATIONS.

109 096 CURIUM 242        NEUTRON                CAPTURE CROSS SECTION
    00  760401              761010    801110
          0.25-1 EV     10.00 MEV    20.0%   3    NFI   M.YADA
        780239  762029   SAFEGUARDS
                     A: 10 PER CENT ACCURACY FOR 25.3 MV, 20 PER CENT ACCURACY FOR HIGHER ENERGY.
                     O: FOR HIGHER BURNUP CALCULATIONS.
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                    ** REQUESTS WITHDRAWN FROM WRENDA.        **                    PAGE   17

110 096 CURIUM 243        NEUTRON                CAPTURE CROSS SECTION
    00  760401              761010    801110
          0.25-1 EV     10.00 MEV    20.0%   3    NFI   M.YADA
        780242  762030   SAFEGUARDS
                    A: 10 PER CENT ACCURACY FOR 25.3 MV, 20 PER CENT ACCURACY FOR HIGHER ENERGY REGION.
                    O: FOR HIGHER BURNUP CALCULATIONS.

111 096 CURIUM 244        NEUTRON                CAPTURE CROSS SECTION
    00  760401              761010    801110
          0.25-1 EV     10.00 MEV    20.0%   3    NFI   M.YADA
        780244  762031   SAFEGUARDS
                    A: 10 PER CENT ACCURACY FOR 25 MV, 20 PER CENT ACCURACY FOR HIGHER ENERGY REGION.
                    O: FOR HIGHER BURNUP CALCULATIONS.
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5. References

1) Muir D.W.: INDC(SEC)-73/URSF, "World Request List for Nuclear Data, WRENDA79/80", (1979).

2) Igarasi S. et al.: JAERI-M 8062, "Japanese List of Requests for Nuclear Data", (1979).

3) Igarasi S. and Narita T.: to be published.

Appendix. Codes of Laboratories.

JAE	Japan Atomic Energy Research Institute
KKU	Kinki University Atomic Energy Research Institute
KTO	Kyoto University
KYU	Kyushu University
MAP	Mitsubishi Atomic Power Industries, Inc.
NAG	Nagoya University
NFI	Nuclear Fuel Industries
NIG	Nippon Atomic Industry Group Co., Ltd.
NIR	National Institute of Radiological Sciences
OSA	Osaka University
PNC	Power Reactor and Nuclear Fuel Development Corporation
SAE	Sumitomo Atomic Energy Industries, Ltd.
TIT	Tokyo Institute of Technology
TKO	Tokyo University
TOH	Tohoku University
TOS	Toshiba Research and Development Center

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