High-Temperature Reactor Physics and Fuel Cycle Studies

HTR-N

Internal Report

„ DESCRIPTION OF CODES APPLIED IN HTR-N“

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PROJECT COORDINATOR: Forschungszentrum Juelich GmbH, D

PARTNERS: ANSALDO ENERGIA S.p.A., I
Commissariat à l'Energie Atomique, F
SUBATECH, F
National Nuclear Corporation, UK
Nuclear Research and Consultancy Group, NL
Joint Research Centre, ITU, D
Delft University of Technology, NL
Universitaet Stuttgart, D
SIEMENS-KWU, D
Universita di Pisa, I

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Code Descriptions

A description of the codes and code packages to be used by the different partners in Work Packages 1, 2 and 3 is given below.

- ***FZJ***

VSOP

This is a system of codes linked together for the simulation of reactor life histories and temporary in-depth research. It comprises neutron cross section libraries and processing routines, repeated neutron spectrum evaluation, 2-D or 3-D diffusion calculation with depletion and shut-down features, in-core and out-of-pile fuel management, fuel cycle cost analysis and thermal hydraulics (the latter at present restricted to HTRs and 2-D). The code system has extensively been used for comparison studies of reactors, their fuel cycles, simulation of safety features, development research, and reactor assessments. It is almost unique for analyses of pebble-bed high temperature reactors in the sense, that it allows the description of all sorts of fuel management that has been discussed for that reactor type up to now, including once-through (OTTO), multi-path (MEDUL) and pseudo batch-load (PEU-A-PEU) fuelling types.The code has been successfully applied to LWR, HWR and hybrid systems as well.

NITAWL-II

Nitawl-II is a module to convert AMPX master libraries to working libraries and to perform resonance self-shielding calculations. The resonance self-shielding calculation is for the resolved energy range and employs the Nordheim Integral Treatment.
This treatment is a two-region integral transport theory method, a fuel lump surrounded by a moderator region, in which the moderator region is assumed to have an asymptotic (1/E) flux everywhere and at all energies. Escape probabilities are used to account for coupling between the two regions. NITAWL uses tabulations of exact probabilities for slabs, cylinders and spheres as developed by Case, de Hoffman and Placzek, but which are based on assuming that the flux in the fuel lump is flat and the scattering is isotropic. To account for lattice effects, a Dancoff factor can be specified, though it is a single value, which is used for all energies of all resonances.

TOTMOS

TOTMOS is a 1-D integral transport code for neutron spectrum calculations and generating homogeneised broad group cross sections, which can be used in detailed reactor calculations.
TOTMOS solves the linear transport equation by use of collision probabilities in a P-0 transport corrected approximation. In order to accelerate the convergence behaviour of the solution method, inner iteration scaling and thermal upscatter scaling are used. Neutron cross section data in the AMPX working library format can be processed.

MARCO POLO

MARCO POLO calculates diffusion coefficients in radial (Dr) and axial (Dz) directions of a cylindric Wigner-Seitz cell using single or multi-group theory. The cell may contain several concentric zones of different materials. Either isotropic or linear anisotropic scattering can be treated.

The diffusion coefficients are calculated assuming zero leakage in radial and axial directions. The obtained diffusion coefficients may be used in reactor calculations with 2-D or 3-D diffusion codes.

CITATION

CITATION is designed to solve problems involving the finite-difference representation of diffusion theory treating up to three space dimensions with arbitrary group-to-group scattering. X-Y-Z, Theta-R-Z, Hexagonal-Z and Triagonal-Z geometries may be treated. Depletion problems may be solved and fuel managed for multi-cycle analysis. Extensive first order perturbation results may be obtained given microscopic data and nuclide concentrations. Statics problems may be solved and perturbation results obtained with macroscopic data.

Explicit finite-difference approximations in space and time have been implemented. The neutron flux eigenvalue problems are solved by direct iteration to determine the multiplication factor or the nuclide densities required for a critical system. Effective techniques are incorporated to determine a critical system. More than one set of microscopic cross sections may be used in a system and nuclide behaviour can be followed on sub-zone scale within depletion regions. The user has flexible control over the route of a calculation as well as the edit of results.

DORT

DORT is a two-dimensional Discrete Ordinates Transport Code. It determines the flux or fluence of particles throughout a two-dimensional geometric system due to sources either generated as a result of particle interaction with the medium or incident upon the system from independent sources. The principal application is to the deep penetration transport of neutrons and photons. Criticality (k-type and search) problems can be solved. Numerous printed edits of the results are available, and results can be transferred to output files for subsequent analysis.

The Boltzmann equation is solved using the method of discrete ordinates, diffusion theory, or a special combined P1 solution. Methods are available to accelerate convergence by space-dependent rebalance and by successive over-relaxation. Anisotropic cross sections can be expressed in a Legendre expansion of arbitrary order. Output data sets can be used to provide an accurate restart of a previous problem.

MCNP

MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron or coupled neutron/photon/electron transport, including the capability to generate Eigenvalues for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori.

Pointwise cross section data are used. For neutrons, all reactions given in a particular cross section evaluation are accounted for. Thermal neutrons are described by both the free gas and $S(\alpha,\beta)$ models.

Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross section data.

- ***CEA***

APOLLO2

This is a lattice code under development at CEA and is currently used for the modelisation of most types of thermal lattices available in France. This is a 2D transport theory program with advanced capabilities for performing self-shielding calculations. It contains 1D and 2D collision probability techniques and discrete ordinates (Sn) capabilities. APOLLO2 also provides interface current approximations which are adapted to model PWR assemblies. The new library CEA93 based on the JEF2.2 evaluation has been completed for APOLLO2. This library is available in two standard multi-group formats: the traditional 99-group energy mesh and the Xmas 172-group structure.

APOLLO2 features a new modular architecture where the data flow is represented in a user oriented supervision language named GIBIANE. This brings a lot of flexibility, as users can implement any calculational scheme in GIBIANE, without having to recompile the code. Thanks to APOLLO2's modular structure and to the commande language GIBIANE, APROC procedural libraries have been constructed and stored to help users run dedicated calculations using the appropriate code options. A modern GUI package named SILENE can help the user to construct complicated calculation meshes in regular and non regular geometries.

Powerful methods for computing multi-group self-shielded cross sections, for homogenizing and collapsing cross sections, heterogeneous-geometry leakage models and modern 2D & 3D transport solvers are available in the code and give it high potentialities and versatility. The general purpose, parametrized output library SAPHYB, organizes all the code results (macroscopic and microscopic cross section data resulting from homogenization and collapsing, fluxes, kinetic parameters, equivalence coefficients, etc.) for an easy access by other code packages such as CRONOS2.

A large number of comparisons with Monte Carlo calculations has been carried out to validate APOLLO2 in a wide range of applications. Also, APOLLO2 participates in many benchmark calculations and confrontation to experimental data is continuously made. Many experiments, realized in the reactors of the CEA or by other organisms around the world (mainly in the critical field) have been and are calculated with APOLLO2.

Thanks to a continuous developmental effort, APOLLO2 has become an outstanding tool for diversified applications covering sophisticated R&D studies that need state-of-the-art methodology, as well as routine industrial calculations for reactor and criticality analysis. APOLLO2 is presently a component of the SCIENCE reactor physics product in use in Framatome.

CRONOS2

This is a core calculation code under development at CEA. It is a highly modular code with varied capabilities: transport calculations with discrete ordinates (Sn) technique, diffusion calculations, detailed isotopical evolution, point kinetics, etc. It uses the SAPHYB libraries produced by APOLLO2 (homogenized and collapsed macroscopic cross sections). It can also simulate the effects of control rod insertion and thermo-hydraulical feed-back.

As for APOLLO2, the data flow in CRONOS2 is represented in the user oriented supervision language GIBIANE. Thanks to this commande language and to the modular structure of the code, CPROC procedural libraries have been constructed and stored to help users run dedicated calculations using the appropriate code options.

TRIPOLI4

This is a Monte Carlo 3D transport code under development at CEA. It uses point-wise cross sections based on the JEF2.2 evaluation. It can perform neutron and gamma transport calculations. TRIPOLI4 is devoted to two main categories of problems : radioprotection (long distance transport with high flux attenuations) and neutronics (critical or sub-critical media).

- ***NRG***

WIMS (Winfrith Improved Multi-group Scheme)

This package consists of a variety of modules to calculate the neutronics for the scheme of cross section generation, cell calculations and pseudo reactor calculations. From a main data library based on the JEF nuclear data file, neutron cross sections can be processed to problem dependant resonance shielded cross sections by means of modules based on collision probability, diffusion theory and differential transport theory. Depletion calculations can be performed in the cell calculations to create burn-up dependent cross section sets, which can be processed to a format to be used by PANTHER.
The WIMS code has been developed by AEA-Winfrith and has been in use for a long time for neutronics calculations for all type of reactors all over the world.

PANTHERMIX (PANther-THERMIX)

This code consists of the two code packages PANTHER and THERMIX together with coding needed to transfer data from the one code to the other and visa versa.
PANTHER is a modular code package designed to solve the Boltzmann equation for neutrons in a thermal reactor, in up to three dimensions, by means of multi-group diffusion theory with a coupled thermal and poison or control rod feedback. It also contains modules to perform the thermal hydraulic calculations but this part is not used for HTR calculations. Temperature and depletion dependent nuclear cross sections are generated by means of the WIMS code package and stored in a massive nuclear data array in PANTHER.
The PANTHER code has been developed by Nuclear Electric Ltd. and has been applied for a variety of thermal reactors like PWR, BWR, AGR, RBMK and MAGNOX.
THERMIX is a two dimensional thermal hydraulics reactor code package allowing for cross flow between the reactor channels. For this reason the PANTHER thermal hydraulic modules are replaced by the THERMIX code to be able to allow for natural convection, which is important for HTR pebble bed reactors. It calculates radiative, conductive and convective heat transfer for a given power distribution and coolant flow.
The THERMIX code, also known under the version DIREKT, has been developed by Forschungszentrum Jülich and has been applied for reactors like the AVR, THTR, and the PBMR.
In the PANTHERMIX package power distributions will be calculated by PANTHER and handed over to THERMIX which calculates a temperature distribution, which is transferred back to PANTHER thus iterating to a converged temperature and power profile for as well stationary as in-stationary reactor states. Provisions have been made to be able to model HTR's with cycling fuel pebbles.
Also the RELAP thermodynamic code package can be coupled to PANTHERMIX to impose the behaviour of the energy conversion system on the reactor coolant inlet flow conditions.
The intervening codes between the different code packages have been developed by NRG-Petten.

CSS1SMAT (Calculate Single burn-up Step 1-group Sensitivity MATrices)

In the assessment of various burn-up and recycling scenarios, it is important to calculate the uncertainties in the final nuclide densities, caused by uncertainties -and covariances- in the basic nuclear data, like multi-group cross sections and decay constants.
Ideally, although still in a linear approximation, a sensitivity matrix of an entire burn-up and recycling scheme would be calculated, which would relate uncertainties (covariances) in final nuclide densities to uncertainties (covariances) in multi-group cross section and decay constant data. As a general burn-up calculation scheme consists of consecutive sets of spectrum-, cross section collapse- and one-group burn-up step calculations, the (relative) sensitivity matrix of the scheme can be constructed out of the (relative) sensitivity matrices of the several consecutive steps. The "multi-group to one-group cross section" sensitivity matrix basically involves the spectrum/weighting function, which is expected to vary from step to step and a "one-group burn-up step".
So, in order to be able to calculate uncertainties and covariances in final nuclide densities resulting from a multi-step spectrum and burn-up calculation scheme, the burn-up part of the calculation scheme must be able to calculate both types of sensitivity matrices mentioned above. Also a third sensitivity matrix is involved; the "decay constant-to-density" sensitivity matrix but is mostly not considered due to a lack of data and on the fact that the influence of the covariances of cross sections on the uncertainties of final nuclide densities is expected to be much larger than that of uncertainties in decay constants.
The burn-up module of the WIMS code package is not able to calculate any of the required sensitivity matrices, whereas the burn-up code FISPACT can e.g. only calculate the sensitivity of the final nuclide densities with respect to uncertainties in the microscopic cross sections and decay constants. Therefore the code CSS1SMAT is being developed at NRG. This code solves the Bateman burn-up equations by assuming a polynomial expansion of the nuclide densities versus time. The code calculates the final nuclide density vector N(i+1) at the end of a given step i (process time ti) assuming a constant one-group flux and constant one-group microscopic cross sections. It can also perform a flux search to obtain a fixed final value of the density of a specified nuclide. This option can be used to simulate a burn-up calculation at constant power.

Versions and References

WIMS :	Version WIMS-8A
JEF:	Version JEF-2.2
PANTHER:	Version PANTHER-5.1
THERMIX:	Version DIREKT-99
RELAP:	Version RELAP-5.0
FISPACT:	Version FISPACT-4.0

➢ ***IRI***

INAS

At IRI, a comprehensive reactor code system is implemented for detailed reactor-core calculations called **INAS** (IRI-NJOY-AMPX-SCALE). This reactor-physics code system uses data based on the JEF-2.2, ENDF/B-VI and JENDL-3.2 evaluated nuclear data file. The code package is used for Validation and Verification (V&V) of the JEF-2.2 evaluated nuclear data file. International benchmarking is performed with NEA DATA BANK (JEF) specifications and for the HTTR reactor in Japan. The code package is also used for reactor-physics calculations and fuel management for the HOR reactor at IRI.

The evaluated data is processed with the well-known cross-section processing code NJOY (97.107) and the NSLINK program that converts data from the NJOY format to AMPX format. The modular code system SCALE (4.2) is used for performing standardised computer analysis for licensing evaluation. This system encompasses unresolved and resolved resonance calculations and cell calculations to generate weighted broad-energy group data for post processing to be used in 1-, 2- and 3-dimensional diffusion and transport codes. The diffusion code BOLD-VENTURE is mostly used for 2-dimensional core calculations. This code can be used for burn-up calculations as well. A special code sequence called SAS6, based on WIMS/D-4 and ORIGEN-S, can be used for detailed cell burn-up calculations. The discrete-ordinates codes from the DOORS package (DORT and TORT) are used for two- and three-dimensional coupled neutron-gamma transport calculations. Also the Monte-Carlo codes MCNP4B, MCNPX and KENO-Va are included in the INAS code system.

- **IKE**

For reactor physics calculations in WP 2 and WP 3 following code packages will be used from IKE:

Codes developed by IKE:

RSYST
is a software tool, that supports an engineer's work in the generation of application programs or program systems, in the execution of programs and in the administration and analysis of data. The main purpose of RSYST is the interconnection of data and application programs. The subsystem RSYST-RP contains application modules for general reactor physics calculations such as solution of 1D – 3D neutron diffusion equation, SN transport equation, first collision transport equation, multi-group cross section processing like condensation, homogenisation etc., spectral calculation, solution of slowing down equation by means of 1D-3D first collision probabilities for cells, burn-up calculations and general purpose modules. In a general data base all data for reactor physics calculations are stored and can be easily taken from every application module. RSYST-RP was applied in former HTR projects performed in Germany and is also applied for load following calculations of LWR, research reactors and fuel cycle studies. For HTR-N, the following RSYST-RP application modules are very important:

DIFGEN	(solution of 1D-3D diffusion equation by finite element method, with special treatment of absorber zones and cavities)
DIFF-H	(solution of 1D-2D diffusion equation by finite difference method with special treatment of cavities)
RESAB	(solution of slowing down equation for arbitrary number of energy points and nuclides in double heterogeneous cells)
RESMOD	Spectral code for generation of few group cross section libraries based on 292 group library in SCALE/AMPX format (JEF-2.2 and ENDF/B-VI)
ISOSTO	First collision transport calculations
ABBRAND	Solution of multi-zone burn-up equations

RESMOD
This is a general spectral program which calculates neutron cell spectra for an arbitrary number of neutron groups with special treatment of the resolved resonance range. The resonance cross sections are pointwise cross sections with 10,000 to 20,000 energy points. An arbitrary number of nuclides can be treated in all zones of the cell. By this method all

important self shielding and mutual shielding effects are taken into account. Double heterogeneity can be treated by corresponding first collision probabilities based on Teuchert's method. The data library used from RESMOD is a 292 group library (N292) generated by NJOY for the main actinides, fission products, moderators, structure materials and absorbers. The data are tabulated for temperatures up to 2500 K and σ0. The library is mainly based on JEF-2.2 and ENDF/B-VI evaluated data. The data format is AMPX, the library is compatible to the SCALE 4.xx modules. The unresolved resonance region is treated by the SCALE module BONAMI (Bondarenko method). The library contains no resonance parameters since the resolved resonance range is treated by solving the slowing down equation based on the pointwise cross sections (comparable to the method in module ROLAIDS of SCALE). The pointwise cross sections of resonance absorbers are also a part of the N292 library (tabulated for several temperatures), but in future there is also a link to the MCNP pointwise cross section libraries which contain also pointwise cross sections for different temperatures. The cross sections which are given for different mesh grids in MCNP library for every nuclide and temperature are interpolated for an unified mesh grid used in RESMOD. The program RESMOD and the corresponding library was extensively tested against experiments (benchmarks) and compared with alternative solutions, especially with solutions with the Monte Carlo code MCNP. For regular cell clusters which can be treated by RESMOD, very good agreement with alternative best estimate solutions could be observed. The agreement with experiments depends not only from quality of method but also from quality of nuclear data.

International available codes applied by IKE:

MCNP-4B2/-4C

MCNP is a general-purpose, continuous-energy, generalized geometry, time-dependent, coupled neutron-photon-electron Monte Carlo transport code system.

MCNP treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise continuous-energy cross section data are used, although multi-group data may also be used. Fixed-source adjoint calculations may be made with the multi-group data option. For neutrons, all reactions in a particular cross-section evaluation are accounted for. Both free gas and S(alpha, beta) thermal treatments are used. Criticality sources as well as fixed and surface sources are available. A very general source and tally structure is available. The tallies have extensive statistical analysis of convergence. Rapid convergence is enabled by a wide variety of variance reduction methods.

MCNP can be extensively used for HTR-N, since transport solutions are possible for a very detailed description in energy and geometry.

DOORS-3.2 package

TORT calculates the flux or fluence of particles due to particles incident upon the system from extraneous sources or generated internally as a result of interaction with the system. TORT is used in two- or three- dimensional geometric systems, and DORT is used in one- or two- dimensional geometric systems. The principle application is to the deep-penetration transport of neutrons and photons. Certain reactor eigenvalue problems can also be solved. Numerous printed edits of the results are available, and results can be transferred to output files for subsequent analysis.

The Boltzmann transport equation is solved using the method of discrete ordinates to treat the directional variable and finite-difference methods to treat spatial variables. Energy dependence is created using a multi-group formulation. Time dependence is not treated.

Starting in one corner of a mesh, at the highest energy, and with starting guesses for implicit sources, boundary conditions and recursion relationships are used to sweep into the mesh for each discrete direction independently. Integral quantities such as scalar flux are obtained from weighted sums over the directional results. The calculation then proceeds to lower energy groups, one at a time. Iterations are used to resolve implicitness caused by scattering between directions within a single energy group, by scattering from an energy group to another group previously calculated, by fission, and by certain boundary conditions. Methods are available to accelerate convergence. Anisotropic scattering is represented by a Legendre expansion of arbitrary order, and methods are available to mitigate the effect of negative scattering estimates resulting from finite truncation of the expansion. Direction sets can be biased, concentrating work into directions of particular interest. Fixed sources can be specified at either external or internal mesh boundaries, or distributed within mesh cells.

SCALE-4.4

The SCALE system was developed for the Nuclear Regulatory Commission to satisfy a need for a standardized method of analysis for the evaluation of nuclear fuel facility and package designs. In its present form, the system has the capability to perform criticality, shielding, and heat transfer analyses using well established functional modules tailored to the SCALE system. Spent fuel characteristics for these analyses can be obtained from a module that performs a depletion/decay calculation. The CSAS control module contains criticality safety analysis sequences that calculate the neutron multiplication factor for one-dimensional (XSDRNPM-S) and multi-dimensional (KENO V.a) system models. The CSAS module also has the capability to perform criticality searches (optimum, minimum, or specified values of k-eff) on geometry dimensions or nuclide concentrations in KENO V.a. The CSAS6 control module contains criticality safety analysis sequences using the KENO-VI module for multidimensional models with more complex geometries, including hexagonal arrays. Sequences that provide problem-dependent cross sections for use in stand-alone codes are also available in the CSAS module. Five shielding analysis sequences are provided. SAS1 analyzes general one-dimensional shielding problems via XSDRNPM-S. SAS3 provides a general procedure for cross-section preparation followed by a shielding analysis using the MORSE-SGC Monte Carlo code. The SAS4 module has been tailored to perform a Monte Carlo shielding analysis for a cask-type geometry. An automated scheme is incorporated in SAS4 to generate Monte Carlo biasing parameters that enable SAS4 to calculate accurate doses with reasonable efficiency. The SAS2H module uses ORIGEN-S to perform a fuel depletion analysis (to characterize spent fuel and/or generate source terms) and an optional one-dimensional radial shielding analysis of a cask-type geometry. ORIGEN-ARP is an automated depletion decay sequence for both PCs and workstations. It includes a menu-driven input processor (MS-DOS PC program) for ORIGEN-S and ARP (Automated Rapid Processing), which automatically interpolates cross sections on enrichment, burn-up, and optionally moderator density using a set of standard basic cross-section libraries for four LWR fuel assembly designs. The interpolated cross sections are passed to ORIGEN-S. Utility codes are provided so users can generate their own ORIGEN-ARP basic cross-section libraries via SAS2H.

SUSD

SUSD calculates sensitivity coefficients for one and two-dimensional transport problems. Variance and standard deviation of detector responses or design parameters can be obtained using cross-section covariance matrices. In neutron transport problems, this code is able to perform sensitivity-uncertainty analysis for secondary angular distribution (SAD) or secondary energy distribution (SED). The first-order perturbation therory ist used to obtain sensitivity coefficients.

DANTSYS-3.0

One-, Two-, and Three-Dimensional, Multi-group, Discrete-Ordinates Transport Code System and includes five major codes. ONEDANT solves the one-dimensional multi-group transport equation in plane, cylindrical, spherical and two-angle plane geometries. TWODANT solves the two-dimensional multi-group transport equation in x-y, r-z and r-theta geometries. TWOHEX solves the two-dimensional multi-group transport equation on equilateral triangular meshes in the x,y plane. THREEDANT solves the three-dimensional multi-group transport equation in x-y-z and r-z-theta geometries. TWODANT/GQ solves the two-dimensional transport equation in x-y and r-z geometries on general quadrilaterals. DANTSYS accepts the basic multi-group cross sections for isotopes, in either of the standard interface files (ISOTXS or GRUPXS) or in a card-image library whose form is referred to as Los Alamos, ANISN, or FIDO. PSR-317/TRANSX-2.15 will translate MATXS libraries into these formats.

ONEDANT, TWODANT, TWODANT/GQ and THREEDANT use discrete ordinates approximation for treating the angular variation of the particle distributions. The diamond difference scheme is used for phase space discretisation. In TWODANT and THREEDANT there is an option to use the adaptive weighted diamond method. Both inner and outer iterations are accelerated using the diffusion synthetic acceleration method. TWOHEX uses the discrete ordinates form for treating the angular variation of the particle distribution, and a nodal scheme is used for phase space discretisation. Both inner and outer iterations are accelerated using the Chebyshev acceleration method.

NJOY-99

The NJOY nuclear data processing system is a modular computer code used for converting evaluated nuclear data in the ENDF format into libraries useful for applications calculations. Because the Evaluated Nuclear Data File (ENDF) format is used all around the world (e.g., ENDF/B-VI in the US, JEF-2.2 in Europe, JENDL-3.2 in Japan, BROND-2.2 in Russia), NJOY gives its users access to a wide variety of the most up-to-date nuclear data. NJOY provides comprehensive capabilities for processing evaluated data, and it can serve applications ranging from continuous-energy Monte Carlo (MCNP), through deterministic transport codes (DANTSYS, ANISN, DOORS), to reactor lattice codes (WIMS, EPRI). NJOY handles a wide variety of nuclear effects, including resonances, Doppler broadening, heating (KERMA), radiation-damage, thermal scattering (even cold moderators), gas production, neutrons and charged particles, photoatomic interactions, self shielding, probability tables, photon production, and high-energy interactions (to 150 MeV).

NJOY consists of a set of modules, each performing a well-defined processing task. The methods and instructions on how to use them are documented in the LA-12740-M report on NJOY91 and in the "README" file. No new published document is yet available.

TRANSX-2

This is a computer code that reads nuclear data from a library in MATXS format and produces transport tables compatible with many discrete-ordinates (SN) and diffusion codes. Tables can be produced for neutron, photon, charged-particle, or coupled transport. Options include adjoint tables, mixtures, homogeneous or heterogeneous self-shielding, group collapse, homogenisation, thermal up-scatter, prompt or steady-state fission, transport corrections, elastic removal corrections, and flexible response function edits. TRANSX reads through the materials in a MATXS library and accumulates the cross sections into a transport table using the user's mix instructions. At the same time, response function edit cross sections are accumulated using the user's edit instructions. They can thus be any linear combination of the cross sections available in the library. When the table is complete, it is written out in the

desired format. Output options include DTF-style card images, FIDO, ISOTXS, and the binary group-ordered GOXS format. Self-shielding is handled using the background cross section method. Heterogeneity options are included.

➢ ITU

Calculations will be done using the **SCALE** modular system (version 4.4). This system was developed at ORNL for the NRC to provide a tool for a standardized method of analysis for the evaluation of fuel facility and transport design. It can perform criticality, shielding and heat transport evaluations. It is licensed and not a single code, but a modular system: that means a collection of computer codes, each one performing a specific task, that can be interconnected thanks to a standardized compatibility of the input/output files. These single codes can be sequentially linked to for calculation sequences defined by the user. The system provides also some pre-defined sequences, called procedures, that allow to generate with a simple condensed input some code sequences required for the most common tasks: like shielding analysis, spent fuel characterization, criticality analysis. The main modules used in this work are:

- **BONAMI** retrieves multi-group cross sections and performs for all the nuclides a preliminary calculation of the self-shielding based on a simplified zero-dimensional method (Bondarenko method);
- **NITAWL** computes for the main resonant nuclides the self-shielding factors using the Nordheim integral method which takes into account the 1-D pin geometry;
- **XSDRNPM** solves the Boltzmann equation of neutron transport in the 1-D cell geometry, computing the space-energy distribution of neutron flux, then computes the cell k-effective and eventually condenses the cross sections;
- **COUPLE** updates ORIGEN libraries with the cross sections and spectral parameters computed by XSDRNPM, creating problem and burn-up dependent libraries;
- **ORIGENS** computes the isotopic evolution of fuel composition;
- **KENO** is a 3-D Montecarlo code to compute k-effective and neutron flux distribution in complex geometry.

Most calculations will be done using the sequences provided in SCALE:

- **CSAS1X** executes BONAMI and NITAWL for cross section treatment and then XSDRNPM to compute the k-effective in simple geometry;
- **CSAS2X** adds to the same sequence of CSAS1X the execution of KENO to allow the treatment of three-dimensional geometry;
- **SAS2H** is the typical iterative sequence used to perform burn-up analyses: the series BONAMI-NITAWL-XSDRNPM is repeated at each time step to create condensed cross sections specific of the cell geometry and fuel composition as a function of burn-up, then ORIGEN computes the time evolution of fuel.

- The **SCALE** system is provided with a wide set of different cross section libraries: for these studies we have selected the 238-group library derived from ENDF/B-V.

- **ANSALDO/UNIPI**

Actual versions of the codes MCNP and DORT or equivalent will be used by ANSALDO and UNIPI. Those codes are already illustrated above.

The compilation of codes shows that the partnership possesses the necessary analytical tools for coping with the programme objectives. The codes will be selected and applied according to the specific needs in the different Work Packages and Tasks.