REVIEW

Development, Validation and Applications of SRAC:
JAERI Thermal Reactor Standard
Neutronics Code System

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The JAERI thermal reactor standard neutronics design code system, SRAC has been developed at JAERI to establish an overall neutronics calculation code system. While incorporating the conventional transport and diffusion codes, SRAC is characterized by application of the collision probability method on the resonance absorption and the cell calculations over the whole neutron energy. A comprehensive set of collision probability routines for 13 types of geometries yields wide application of SRAC to almost all types of thermal reactors. Since the functions have been qualified through a wide range of benchmark calculations, SRAC has been used for neutronics design of the core conversion of JAERI research and test reactors and the analyses of critical experiments performed at JAERI and domestic universities. Recent modifications of methods and data extend its applicability to HCLWRs cores.

KEYWORDS: neutronics, code system, cell calculations, resonance absorption, collision probability method, benchmarks, experimental data, thermal reactors, SRAC

I. INTRODUCTION

The development of the SRAC code system\(^{(1)}\) was started in 1978 to share the neutronics part of the JAERI thermal reactor standard code system. Demands were growing for more accurate estimate of reactor characteristics, safety aspects and fuel cycle strategies, at that time when the core conversion of the JAERI research and test reactors for the use of reduced enrichment fuel, the conceptual design of a test facility of HTGR, and the utilization of plutonium in the light water power reactors were the urgent programs.

The system has been designed to permit the application to a wide range of reactor types for a variety of usages such as a feasibility study, a conceptual design and an experimental analysis. To fulfill the purpose, many options have been implemented for selection of methods, energy group structures and geometries.

The applications of the primary version of the SRAC code system\(^{(3)}\) were started in 1981 after the verification by extensive benchmark calculations. An increase of users has accelerated the debugging of many unexperienced functions since then.

Effort has been continued to establish an overall neutronics code system. In addition to the cross section library derived from ENDF/B-4, the JENDL-2 version has also been available. An auxiliary code for the core burnup and the fuel management has been incorporated, and a number of additions and modifications of the functions have been made.

II. MAIN FEATURES OF SRAC

The SRAC code system is designed to permit overall neutronics calculation which
covers microscopic cross section library compilation, macroscopic constant generation, cell and core calculations including the burnup and the fuel management. The kinetics and safety related parameters are also provided. The above functions are explained in Fig. 1.

The unique features implemented in SRAC are described as follows:

1. Flexible Energy Group Structure
   The energy group structure of the fundamental cross section libraries is sufficiently fine. For the computer time and storage saving “User” libraries for the fast and thermal group constants are composed to install only the data of the relevant nuclides in a user’s multi-group structure. After collapsing is made by using the built-in asymptotic spectrum, the macroscopic cross sections are composed.

   The few group structure is also specified by the user.

2. Optional Cell Calculation
   Several kinds of optional modules are available for the cell calculation. The module based on the collision probability method can treat 13 types of geometries. The SN module utilizes the ANISN code(9) for 1D calculation and the TWOTRAN code(4) for 2D.

   The multi-group cell calculation can be performed separately in each energy range of fast and thermal neutrons as a fixed source problem, or at once through the whole neutron energy range as an eigenvalue problem or as a fixed boundary source problem.

3. Wide Reactor Types to be Applied
   A comprehensive set of collision probability routines for various lattice geometries permits the cell calculation of a variety of thermal reactors. The list of geometries SRAC can treat is given in Table 1.

4. Optional Treatment for Resonance Absorption
   Three options for the treatment of the resonance absorption are available. The effective cross sections by the conventional table-look-up method based on the NR approximation can be replaced by those based on the

![Flow diagram of SRAC](image)

**Fig. 1** Flow diagram of SRAC

**Table 1** Lattice geometries treated by collision probability method

<table>
<thead>
<tr>
<th>No.</th>
<th>Geometry</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>1/2/3</td>
<td>1D sphere/slab/cylinder</td>
<td>Multi-layer</td>
</tr>
<tr>
<td>4/5</td>
<td>Square cylinder with annular/2D division</td>
<td>BWR type</td>
</tr>
<tr>
<td>6/7</td>
<td>Hexagonal cylinder with annular/2D division</td>
<td>ATR type</td>
</tr>
<tr>
<td>8</td>
<td>Square assembly with X-Y array of rods</td>
<td>HTGR type</td>
</tr>
<tr>
<td>9</td>
<td>Annular assembly with annular array of rods</td>
<td>HTGR type</td>
</tr>
<tr>
<td>10</td>
<td>Annular assembly with asymmetric array of rods</td>
<td>HCLWR type</td>
</tr>
<tr>
<td>11</td>
<td>Hexagonal assembly with asymmetric array of rods</td>
<td>FBR type</td>
</tr>
<tr>
<td>12</td>
<td>Triangular assembly with two rod type</td>
<td></td>
</tr>
<tr>
<td>13</td>
<td>Hexagonal assembly with triangular array of rods</td>
<td></td>
</tr>
</tbody>
</table>
IR approximation. Also provided is a direct method to solve a multi-region cell problem in a ultra-fine group by the collision probability method. The use of an interpolation scheme for the collision probabilities reduces the computing time to the same order as the table-look-up method.

(5) Treatment of Double Heterogeneity

Smearing to have the homogeneous equivalent cross sections of the cell and collapsing them into the few group structure are performed in separate steps. This separation of smearing from collapsing enables us to treat a highly heterogeneous geometry by successive cell calculations.

(6) Installed Dancoff Correction Factor Calculation

The Dancoff correction factor is optionally calculated by the installed collision probability module. Recent modification introduced a generalized Dancoff correction given not for an absorber lump but for each constituent nuclide to treat so-called “two rod heterogeneity”.

(7) Option for Core Calculations

The core calculation is performed by the SN module mentioned before or by the diffusion module in which the diffusion code CITATION is used.

A preceding multi-group core calculation in a simplified geometry may be used to provide space-dependent few group constants for the core calculation on the more realistic geometry.

Several modifications are made to the CITATION module to yield the kinetics and safety related parameters $\beta_{\text{eff}}$ and $\lambda$, the direction dependent diffusion coefficients, and the material-dependent fission spectrum and delayed neutron fraction.

(8) Burnup Calculation

In the SRAC code system, the burnup process is divided into two steps. First, the cell burnup process yields few group macroscopic cross sections in which the effect of neutron spectrum change is implicitly included. They are prepared on the discrete values of burnup, fuel temperature and coolant void fraction for a cell.

An auxiliary program COREBN to execute 2D or 3D core burnup utilizes this tabulation of few group macroscopic cross sections. The diffusion routine calculates the power distribution to give increased burnup of each spatial node. The changes of composition and also neutron spectrum during burnup are expressed by the changes of macroscopic cross sections which are provided by an interpolation scheme of the macroscopic cross sections.

Information before initial and after final step of burnup is read/written from/to a “History” file which keeps the information of each individual fuel element. It is utilized for the fuel management.

(9) Data Storage on PDS File

The storage and search of a variety of data are carried out by using a kind of PDS (Partitioned Data Set) files. An assembler routine PDSFUTY permits the file/member control (open, close, read, write, delete) by the Fortran statement. The built-in FACOM TSS terminal commands are also available for file control before and after the execution.

III. VALIDATION AND APPLICATIONS

For validation purpose, extensive benchmark calculations were carried out on various types of critical assemblies such as TCA*, DCA, SHE, JMTRC HEU cores. A series of benchmark analyses of TRX and ETA cores showed the good prediction on not only the multiplication factors but also the spectrum parameters such as $\rho_{25}$, $\delta_{28}$, $C^*$. Some examples of the results are given in Table 2. A series of FBR benchmark calculations gave confidence on the adequacy of fast energy group constants.

An international RERTR program has offered occasions to show the validity of SRAC. They are an intercomparison of benchmark calculations for a heavy water moderated research reactor, an analysis of the initial LEU core of FNR, analyses of the temperature and void coefficients of KUCA MEU cores, and a series of analyses of the critical experiments at JMTRC MEU core. Another international benchmark for a BWR lattice with adjacent Gd pins will be explained in ABBREVIATIONS given after Chapter IV.
showed that SRAC can treat the burnup of a highly heterogeneous lattice cell.

The SRAC code system was applied to the designs and safety analyses of the upgrading of the JRR-3\textsuperscript{(43)}, the core conversion of JMTR and JRR-2 for the use of MEU fuel, and the reconstruction of SHE\textsuperscript{(43)} into VHTRC. The benchmark study of the criticality safety facilities\textsuperscript{(20)} may qualify the design and safety analysis of NUCEF.

Table 2 \(C/E\) values\textsuperscript{1} for \(K_{\text{eff}}\) and lattice parameters

<table>
<thead>
<tr>
<th>Assembly</th>
<th>Parameter</th>
<th>Exp.</th>
<th>(C/E)</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>TRX-1\textsuperscript{1\dag}</td>
<td>(K_{\text{eff}})</td>
<td>1.00</td>
<td>0.9993</td>
<td>H/(^{235}\text{U}=250)</td>
</tr>
<tr>
<td></td>
<td>(\rho_{25})</td>
<td>1.311 ± 0.02</td>
<td>1.001</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(\delta_{25})</td>
<td>0.0914 ± 1.056</td>
<td>1.094</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(C^*)</td>
<td>0.792 ± 0.008</td>
<td>0.986</td>
<td></td>
</tr>
<tr>
<td>TRX-2\textsuperscript{1\dag}</td>
<td>(K_{\text{eff}})</td>
<td>1.00</td>
<td>0.9945</td>
<td>H/(^{238}\text{U}=430)</td>
</tr>
<tr>
<td></td>
<td>(\rho_{25})</td>
<td>0.83 ± 0.015</td>
<td>0.983</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(\delta_{25})</td>
<td>0.0608 ± 0.0007</td>
<td>0.965</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(\delta_{25})</td>
<td>0.0667 ± 0.002</td>
<td>1.032</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(C^*)</td>
<td>0.644 ± 0.002</td>
<td>0.985</td>
<td></td>
</tr>
<tr>
<td>DCA</td>
<td>(K_{\text{eff}})</td>
<td>1.00</td>
<td>1.0021</td>
<td>Pitch=22.5 cm</td>
</tr>
<tr>
<td></td>
<td>(\rho_{25})</td>
<td>0.84</td>
<td>0.94</td>
<td>UO(_2) initial core</td>
</tr>
<tr>
<td></td>
<td>(\delta_{25})</td>
<td>0.06</td>
<td>1.00</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(\delta_{25})</td>
<td>0.049</td>
<td>1.00</td>
<td></td>
</tr>
<tr>
<td>ETA-1</td>
<td>(K_{\text{eff}})</td>
<td>1.00</td>
<td>0.9949</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(\rho_{25})</td>
<td>10.54 ± 0.15</td>
<td>0.921</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(\delta_{25})</td>
<td>1.74 ± 0.002</td>
<td>1.012</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(\delta_{25})</td>
<td>0.0166 ± 0.0009</td>
<td>0.842</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(C^*)</td>
<td>0.867 ± 0.009</td>
<td>0.896</td>
<td></td>
</tr>
<tr>
<td>SHE-8</td>
<td>(K_{\text{eff}})</td>
<td>1.00</td>
<td>0.9949</td>
<td>C/(^{235}\text{U}=287) homogenous core</td>
</tr>
<tr>
<td>SHE-13</td>
<td>(K_{\text{eff}})</td>
<td>1.008</td>
<td>1.0031</td>
<td>C/(^{238}\text{U}=5976) heterogeneous core</td>
</tr>
</tbody>
</table>

| Average of \(K_{\text{eff}}\) | 0.9974 |
| S.D. of \(K_{\text{eff}}\) | +0.0034 |

\(\rho_{25}, \rho_{25}\): Ratio of epithermal to thermal \(^{235}\text{U}\) or \(^{232}\text{Th}\) capture.
\(\delta_{25}\): Ratio of epithermal to thermal \(^{233}\text{U}\) fission.
\(\delta_{25}, \delta_{25}\): Ratio of \(^{235}\text{U}\) or \(^{233}\text{Th}\) fission to \(^{233}\text{U}\) fission.
\(C^*, \text{CR}^*\): Ratio of \(^{235}\text{U}\) or \(^{233}\text{Th}\) capture to \(^{235}\text{U}\) fission.
\dag Data in Table 2 are cited from Ref. (8).
\textsuperscript{1} "Cross section Evaluation Working Group Benchmark Specifications", ENDF-202 (BNL-19302), BNL.

Prediction of reactivity coefficients has been validated through analyses of the Doppler effect of coated particle fuel for HTGR\textsuperscript{(24)}, the temperature coefficient of a light water moderated research reactor\textsuperscript{(25)} and the reactivity coefficients of the Chernobyl reactor\textsuperscript{(26)}.

Recent effort on development of HCLWR has led some modifications\textsuperscript{(6)(47)(48)} of method and data to apply SRAC to an intermediate spectrum system. Fairly good agreement shown in an analysis\textsuperscript{(29)} of high conversion LWR experiments performed at the Proteus reactor was obtained by a new collision probability routine to treat "two rod heterogeneity". Analyses of critical experiments for HCLWR performed at KUCA\textsuperscript{(30)} and at FCA\textsuperscript{(31)} show good applicability to HCLWR cores.
IV. CONCLUDING REMARKS

The wide applicability and sufficient accuracy of SRAC mentioned above are fully utilized for analyses of critical experiments, neutronics designs of research and test reactors and feasibility studies of new type reactors at JAERI and the domestic universities.

An innovative plan is started at JAERI to establish a reactor design code system by combining SRAC with some thermohydraulics and structure design codes using an object oriented language. An artificial intelligence technique coupled with a knowledge base will support the usage and extend the utility of the new system.

[ABBREVIATIONS]

TCA : Tank-type Critical Assembly for light water reactor
DCA : Deuterium Critical Assembly for a pressure tube type reactor at PNC Oarai
SHE : Semi-Homogeneous Experimental facility loaded with 20% enriched uranium, moderated by graphite
FCA : Fast Critical Assembly
JMTR : Japan Material Testing Reactor
JRR-2 : Japan Research Reactor-2
JRR-3 : Japan Research Reactor-3
JMTRC : Critical facility for JMTR
RERTR : Reduced Enrichment of Research and Test Reactor Fuels
HEU : Highly Enriched Uranium
LEU : Low Enriched Uranium
MEU : Medium Enriched Uranium
FNR : Ford Nuclear Reactor at Michigan Univ.
KUCA : Kyoto Univ. Critical Assembly
UTR KINKI : University Training Reactor at Kinki Univ.
VHTRC : Very High Temperature Reactor Critical Facility
HTGR : High Temperature Gas-cooled Reactor
HCLWR : High Conversion Light Water Reactor
NUCEF : Nuclear Fuel Cycle Safety Engineering Facility

REFERENCES

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