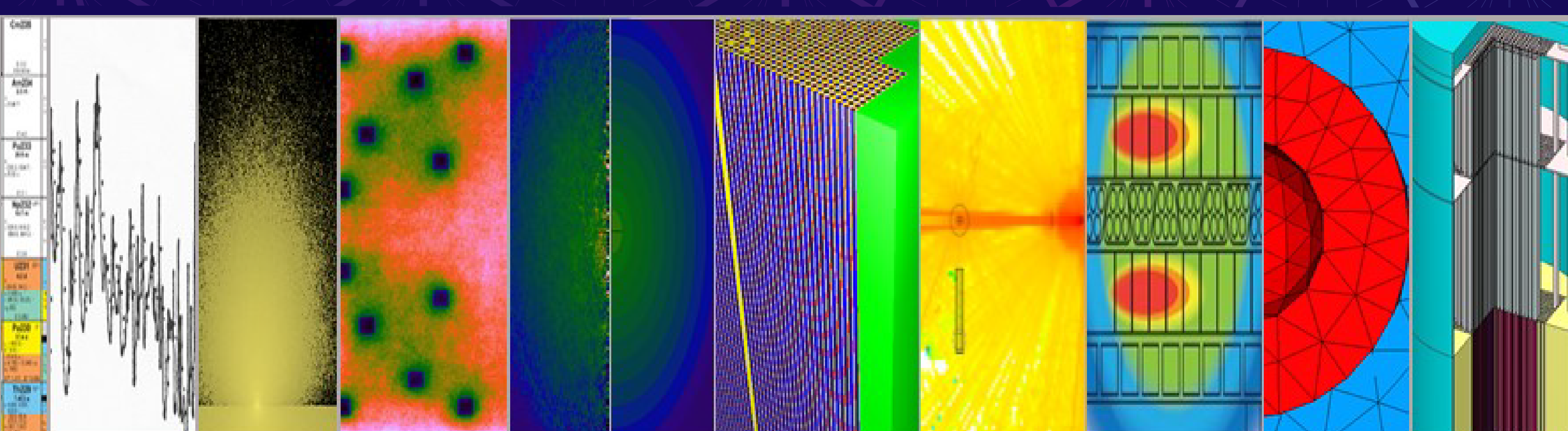
LA-UR-18-20808

MCNP®

Release Notes



Code Version 6.2

MCNP Version 6.2

Release Notes

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Contents

1. Introduction

Monte Carlo N-Particle or MCNP®[1] is a general-purpose Monte Carlo radiation-transport code designed to track many particle types over broad ranges of energies. This MCNP Version 6.2 follows the MCNP6.1.1 beta [1] version and has been released in order to provide the radiation transport community with the latest feature developments and bug fixes for MCNP. Since the last release of MCNP major work has been conducted to improve the code base, add features, and provide tools to facilitate ease of use of MCNP version 6.2 as well as the analysis of results. These release notes serve as a general guide for the new/improved physics, source, data, tallies, unstructured mesh, code enhancements and tools. For more detailed information on each of the topics, please refer to the appropriate references or the user manual which can be found at http://mcnp.lanl.gov. This release of MCNP version 6.2 contains 39 new features in addition to 172 bug fixes and code enhancements. There are still some 33 known issues the user should familiarize themselves with (see Appendix).

2. New Features

MCNP version 6.2 contains 39 new or improved features and utilities that are summarized in Tables 2-1 and 2-2. This section provides a brief description of these features and utilities by category.

TABLE 2-1 New Features and Code Enhancements for MCNP version 6.2

Category	Feature
Physics	• Correlated prompt fission neutron and gamma-ray emission models (CGMF & FREYA) • Improved correlated prompt secondary particle production (CGM) • Exact line emission treatment for delayed gamma production • Decay emission treatment • Charged particle delta-ray production
Source	• Addition of spontaneous positrons decay sources • All-Particle Decay Option

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Category	Feature

	• Spontaneous decay and activation improvements
	• Improved cosmic-ray source:
	  o Inclusion of heavy ions
	  o Updated solar modulation data
	• Improved background source:
	  o Updated cosmic and terrestrial background data file
	  o Automatic elevation and data scaling

Data	• Revised nuclear data for hydrogen • SiO_2 S(α,β) thermal scattering data updated • Zr-Hydride S(α,β) thermal scattering data updated at 1200K • New listing of available ACE data • New default XSDIR file for MCNP version 6.2 • New electron-photon relaxation library (EPRDATA14) • Large-angle and total electron elastic cross section data • Improved decay library data file, increasing radionuclides from 979 to 3475
Tallies	• Built-in physics-based neutron and photon response functions • Improved first-fission special tally option • New collision based cell-flag tally option • Surface flux tally improvements
Unstructured Mesh	• Improved tracking of charged particles on unstructured mesh • Creation of new UM input file type (MCNPUM) • Selection of overlap model by part • Ability to specify flux multipliers on the UM edits • Ability to read and handle multiple UM's in separate mesh universes
Code Enhancements	• Filenames used by MCNP may now be up to 256 characters in length • Permit line lengths up to 128 chars in MCNP input files and xsdir files • Extend command line length to permit up to 4096 characters • Remove limit on boundary-list entries for cell descriptions • The number of point detectors allowed increased from 100 to 1000

TABLE 2-1 New Features and Code Enhancements for MCNP 6.2

Category	Utility
Tools	• Whisper – software for sensitivity-uncertainty-based nuclear criticality safety validation • The MCNPTools package • Intrinsic Source Constructor (ISC) • The UM_CONVERT utility program • Improved UM_POST_OP functionality

2.1 PHYSICS

2.1.1 Correlated prompt fission neutron and gamma-ray emission models (CGMF & FREYA):

Two new correlated fission event generators, CGMF and FREYA, have been integrated into the code to address needs within the nuclear nonproliferation and safeguards communities for high fidelity models of the neutron and gamma-ray emissions from both spontaneous and neutron-induced fission processes [2]. The ultimate use of these models, currently under active development at LANL and Lawrence Livermore National Lab/Lawrence Berkley National Lab, respectively, is to provide a predictive capability in simulating the unique signatures of special nuclear materials in situations where multiple detectors may be used in time-coincidence resulting in correlated counts from fission events. The new fission models can only be used for fixed-source calculations and are turned on with the FMULT card using the METHOD keyword.

2.1.2 New Improved correlated prompt secondary particle production (CGM):

The MCNP library physics treatment suffers from uncorrelated secondary-particle production due to the sampling of inclusive data in the ACE libraries. While the average particle production is preserved, coincidence physics and scoring is not possible. For example, a neutron can undergo an elastic scatter and yet produce a capture gamma. A library-based remedy to this problem is not very practical as the file size necessary to describe exclusive interaction and secondary-particle production cross sections is prohibitive. New in this release of MCNP, a modeled physics solution is provided for secondary-gamma & neutron production via a link to the LANL-developed Cascading Gamma-ray Multiplicity (CGM) code [3]. By setting the 9th entry on the PHYS:N card to “2”, correlated prompt secondary gammas and neutrons are generated by CGM, overriding the ACE table secondary particle production cross sections.

2.1.3 Exact line emission treatment for delayed gamma production:

Delayed or decay-gamma production via a line-emission treatment, with its related *cingergl.dat* data file includes both a source option (PAR=sp on the SDEF card) and an activation option (FISSION/NONFISS=p with DG=line on the ACT card). This capability was based on a mini-bin treatment that involved lumping all lines within each mini-bin (i.e., 25-keV interval) together. This occasionally resulted in oversampling of low amplitude emission lines near a prominent peak. With this new release of MCNP, an exact line-emission treatment was developed [4] that stores all line emission data in a line-by-line cumulative distribution function from which line emissions are directly sampled. Also included in this release, the delayed-gamma energy-biasing algorithm (DGEB keyword on the ACT card) was rewritten to support the new exact line-emission treatment [5].

2.1.4 Decay emission treatment:

MCNP version 6.2 includes an improved time-integration algorithm for spontaneous decay sources, as well as a decay-gamma spectrum improvement for both spontaneous and activation decay [6]. The latter improvement includes gammas specified as a continuum distribution and discrete x-ray lines, which were previously ignored. The continuum data often describes the high-energy component of a spectrum while the x-ray data describes the low-energy component. In regards to the time integration, when a spontaneous-decay source is specified (PAR=sn, sp, sb, sa, st on the SDEF card), an improved time-integration algorithm is invoked which resets all decay constants to unity so as to significantly improve the time integration accuracy. This approach has been found to reduce integration errors by nearly an order of magnitude (in some cases, this error approached 100% and is now within 10% of the expected value).

2.1.5 Charged-Particle Delta-Ray Production:

The production of knock-on electrons, or delta-rays, has been included in the MCNP electron transport for many decades. In this release, we offer a physics option to turn on delta-ray production for heavier charged particles [7], via the 17^{th} entry on the related PHYS card(s). By default this value is “0” which disables production. The user can either set this entry to “-1” to use the default minimum delta-ray production threshold of 0.02 MeV or enter a user specified value of the production threshold.

2.2 SOURCES

2.2.1 Addition of spontaneous positron decay sources:

The decay-particle production capability is extended for this release to include decay-positron emission [8], via the PAR=st option on the SDEF card and/or the FISSION/NONFISS=f option on the ACT card (note that “f” is the official MCNP particle symbol for positrons, but the PAR option “sf” was already taken for spontaneous-fission prompt neutron emission).

2.2.2 All-particle decay option:

The spontaneous-decay source option was extended to provide an all-particle decay option (PAR=sd on the SDEF card) [8], which automatically generates the correct number of all types of source decay particles if those particle types are included on the MODE card. Previously, a user had to specify a distribution of such particle types and adjust the source normalization accordingly.

2.2.3 Spontaneous decay and activation improvements:

Decay-particle production (i.e., ACT card) has been improved for this version of MCNP [4]. The THRESH keyword, which filters out low-probability line emissions, is now applied to the line data of an entire precursor decay chain rather than on a daughter-by-daughter basis; the PECUT keyword was added to enable the omission of gamma lines below a specified energy threshold; and the HLCUT keyword was added to enable the truncation of decay chains based on a specified half-life threshold. The first two options can greatly reduce memory requirements when the DG=line treatment is invoked.

2.2.4 Improved cosmic-ray source:

2.2.4.1 INCLUSION OF HEAVY IONS:

The initial cosmic-source feature (PAR=ch, ca, cr on the SDEF card) included the automatic production of protons and alphas using analytic interplanetary primary spectra, location-dependent rigidity cutoffs, and solar modulation effects. This capability has been extended to include heavy-ion production whenever the user includes heavy-ion transport on the MODE card [9]. These heavy ions are currently represented by Ni, Si, and Fe, appropriately weighted to represent 3 higher-Z GCR bands.

2.2.4.2 UPDATED SOLAR MODULATION DATA:

The automatic solar modulation scaling (i.e., date scaling) was initially implemented using yearly modulation potentials for 1960-2005, with sinusoidal extrapolation before or after these years. In this release of MCNP, we have updated the modulation potentials with the latest data from I. G. Usoskin, which covers the period from 1936-2014 [10]. Two additional improvements include: (1) an algorithm that automatically updates the sinusoidal fitting parameters when subsequent modulation data is added to the MCNP source code; and (2) an automatic sampling technique that determines geospatial-dependent flux-to-current ratios used to adjust the J. M. Clem source normalization (previously, a single fixed ratio of 1.6 was used).

2.2.5 Improved background Source:

2.2.5.1 UPDATED COSMIC & TERRESTRIAL BACKGROUND DATA FILE:

The MCNP *background.dat* file contains neutron and photon spectra at various locations on Earth. Release 1 of this file was produced in 2011 with subsequent releases every year or two. The cosmic-source capability (see section 2.2.4) was combined with terrestrial radionuclide source modeling to generate ground/sea-level neutron and photon background fluxes on a 2^{o} by 10^{o} latitude/longitude grid, and have been incorporated into the improved Release 4 of the *background.dat* file [9]. These spectra can be automatically sampled as a source using the background-source option. Improvements over Release 3 data include a refined geospatial grid, an improved cosmic source, magnetic-field effects, and location-dependent terrestrial photon flux component (within US).

2.2.5.2 AUTOMATIC ELEVATION & DATE SCALING:

An automatic elevation scaling factor has been developed and implemented to apply neutron and photon flux corrections for elevation mismatches between the user-specified elevation and the nearest grid-point elevation [11]. Previously, such corrections had to be calculated and supplied by the user, often leading to significant errors in source normalization. This automatic scaling feature compares the elevation of the nearest grid-point location to that specified by the 3rd entry of the LOC keyword on the SDEF card, and if there is a mismatch, an exponential correction is applied using the scaling factor formulation given in [11] and [12]. In a similar fashion, date scaling is now automatically applied if the date specified on the DAT keyword of the SDEF card differs from the date associated with the background spectra contained in the *background.dat* file [12]. This latter scaling is omitted if the DAT keyword is not specified.

2.3 DATA

The ENDF/B-VII.1 nuclear data files in ACE format are included, along with continuous S(α,β) thermal scattering data, and all older data files are also available. A few datasets in ENDF/B-VII.1 were fixed from previous releases and described as follows.

2.3.1 Revised Nuclear Data for Hydrogen:

The ACE data files for hydrogen based on ENDF/B-VII.1 data that were released with MCNP6.1

and MCNP6.1.1 did not include data for photon production. The ACE files are 1001.80c through 1001.86c. While (n,γ) reactions were properly included in all relevant cross-sections, the specific data for the number and energy of photons produced in the (n,γ) reactions was not included in those ACE files. The updated ACE files for hydrogen are designated 1001.90c through 1001.96c. The neutron cross section data are identical to the previous hydrogen data files, except that the photon production data is included [13].

2.3.2 SiO_2 S(α,β) Thermal Scattering Data:

The SiO_2 S(α,β) thermal scattering data released with MCNP6.1 and MCNP6.1.1 was incorrect due to errors in the ENDF/B-VII.1 data at the time. The ENDF/B-VII.1 errors were corrected and the ACE files for SiO_2 S(α,β) thermal scattering data were regenerated. The previous data, ACE files sio2.30t through sio2.36t, are incorrect and should not be used. The new replacements, sio2.10t through sio2.16t, should be used instead [13].

2.3.3 Zirc-Hydride S(α,β) thermal scattering data at 1200K:

The ACE file for thermal scattering in hydrogen at 1200K released with MCNP6.1 and MCNP6.1.1 was incorrect. The errors were corrected and a new data file is included with MCNP version 6.2. Specifically, ACE file h-zr.27t is incorrect and should not be used; ACE file h-zr.28t is the replacement with corrected data [14].

2.3.4 New listing of available ACE data:

The report listing all of the ACE datasets available with MCNP version 6.2 was updated. This reference should be used in place of previous listings to ensure that the proper ACE data files are used in all calculations [15].

2.3.5 New default XSDIR file for MCNP 6.2:

The XSDIR file used by MCNP is a data file containing available ACE files with the preferred (default) files listed first. Versions prior to MCNP6 used a file named *xsdir*. MCNP6.1 and MCNP6.1.1 used a file named *xsdir_mcnp6.1*. MCNP version 6.2 uses a file named *xsdir_mcnp6.2*. While the *xsdir_mcnp6.2* file can be used with any of the MCNP6 versions, it should not be used with MCNP5, since the default thermal scattering treatment is continuous (which was not correctly handled by MCNP5).

2.3.6 New electron-photon relaxation data:

The MCNP version 6.2 release includes a new and improved Electron-Photon-Relaxation library, called EPRDATA14. This new library is based on data from EPICS2014 [16] that corrects a known deficiency in the electron angular distribution from large-angle elastic scattering, and also includes tabulations of a total electron elastic cross section, so that both the large-angle and the forward-peak angular regions can be sampled in detail. Users interested in the single-event method for electron transport should select the new data by specifying the ID-string ".14p" in material descriptions rather than the now-obsolete data selected by ".12p".

2.3.7 Large-angle and total electron elastic cross sections:

The code and data for the earlier releases MCNP6.1 and MCNP6.1.1 considered only the large-angle electron elastic process (meaning elastic scattering into angles greater than 1.0e-06 in the cosine). Scatter into angles closer to the forward peak was ignored. For many applications this is likely to be an adequate approximation, perhaps failing only when detailed resolution of the forward peak is needed. With EPRDATA14 and new methods in MCNP version 6.2 both the large-angle and the total elastic scattering can be treated. This choice is put under user control using the 13th entry of the PHYS:E card, which now sets the mode of electron elastic scattering as follows:

If PHYS:E (13th entry) = 0, then large-angle elastic scattering is used (default).

If PHYS:E (13th entry) = 2, then total elastic cross section (large-angle + in-peak) is used.

2.3.8 Improved decay library data files:

The decay production data file (*delay_library.dat*) was updated based on data from ENDF/B-VII.1 [17], and includes, for the first time, delayed-positron data. The upgrade to ENDF/B-VII.1 increased the number of decay-neutron radionuclides from 279 to 298, decay-beta radionuclides from 1201 to 1891, and decay-alpha radionuclides from 171 to 248. Plus, the number of gamma bins increased from 25 to 500. In addition, the decay-gamma line data file, *cindergl.dat*, was updated to include all ENDF/B-VII.1 data, increasing the number of radionuclides in that file from 979 to 3475.

2.4 TALLIES

2.4.1 Built-in physics-based neutron and photon response functions:

The built-in detector response functions (FT PHL and DF cards) were expanded to include some neutron scintillation detectors (Li glass, LiI, and ZnS/LiF), as well as production response functions for gas (^{3}He, BF_3) and semiconductor (HPGe) detectors [18]. Since Birks' Law is known to be accurate for Z<6 and E<50 MeV/amu, the use of fitting coefficients were expanded for scintillation production by light ions (Z<4). For the lithium-based scintillators, the coefficients were optimized for triton and alpha energy deposition, although the scintillation from electrons and other light ions (e.g., protons, deuterons) is also treated. The neutron response functions for gas detectors involve the product of the ion energy deposition divided by the gas ionization work function (i.e., w-value in MeV/ion pair), the detector multiplication (M), and the charge per ion pair (in pico-Coulombs). The semiconductor response function for HPGe is treated in a similar manner, only the w-value is ~10 times smaller and the default gain is set to unity.

2.4.2 Improved First-Fission Special Tally Option:

The first-fission special tally option (FFT keyword on the FT card) enables the segregation of a tally based on which actinide first underwent fission within a particle history. The initial implementation included only neutron-induced fission for library-based interactions. This release extends this treatment for all incident particle types (i.e., photon, proton, etc.) and for both model and library-based interactions. Users can turn on/off this tally-flagging feature for each physics treatment via a new FFT input entry.

2.4.3 Improved Collision-Based Cell-Flag Tally Option:

The Cell-Flag tally option (CF card) enables the segregation of a tally based on which cell(s) the tracked particle has entered and has been available in MCNP for decades. In this release of MCNP, this treatment has been extended so as to set the cell flag only when the tracked particle has collided within the specified cell. This extension is invoked only when the user specifies a negative cell number on the CF card.

2.4.4 Surface Flux Tally Improvements:

Surface flux tallies include the inverse of the surface crossing direction cosine. For grazing crossings, where the direction cosine is close to zero, the score can be unbounded, leading to infinite variance. To mitigate this effect, one simply applies an average over the grazing range (as determined by the code or user). Previously, grazing angles are assumed to be those where the dot product of the particle direction and the surface normal are between -0.1 and 0.1. The new default in MCNP was modified such that the grazing angles are now defined between -0.001 and 0.001. Additionally, users can set entry 24 of the DBCN card to a positive number (DBCN(24) > 0), to select a specific grazing angle cutoff.

2.5 UNSTRUCTURED MESH

Since the last release there has been significant work completed in support of the unstructured mesh (UM) feature. This work includes:

2.5.1 Improved tracking of charged particles on unstructured mesh:

The top-level tracking routines for electrons and heavier charged particles (i.e., protons) are separate from each other and quite a bit different from the top-level tracking routine for low-energy neutrons and photons. The different physics for the various particle types is the primary reason for this. There is some level of complexity when implementing this tracking with the UM compared to the Constructive Solid Geometry (CSG) and it has to do with the granularity of the UM. The CSG tracking routines do a good job in terms of knowing what CSG cell the charged particle occupies at all times. UM tracking routines do the same job by knowing what cell (or pseudo-cell) and element in that pseudo-cell the particle occupies. (Note, at this time, all elements in the same part are assigned the same material.) This implies that many more boundary crossings must be taken into consideration. Consequently, over the years of development for the UM feature much work has occurred with the UM charged particle tracking to ensure correctness. This includes proper treatment of charged particle energy deposition edits [19].

2.5.2 Creation of new file type (MCNPUM):

For this version of MCNP, a new filetype is created (MCNPUM) [20] that contains information about the UM internal data structures. An Abaqus input file contains some basic information about the UM, but does not contain everything that MCNP needs. Once MCNP reads this file, it uses the Abaqus data to generate other information that it needs in its tracking routines such as nearest neighbor lists. Even with the parallel input processing, significant computer time can be required to regenerate this data and create other internal data structures for every MCNP calculation that uses the Abaqus unstructured mesh file.

The MCNPUM filetype was created to contain all of the unstructured mesh data structures that MCNP version 6.2 needs, thus eliminating the need to "input process" the Abaqus input file every time the code is run, including continue runs. MCNP can generate this file (primarily after processing the Abaqus input file) by simply including the MCNPUMFILE option on the embed card. MCNP can use the MCNPUM file when the MCNPUM keyword is supplied to the MESHGEO parameter on the EMBED card.

Note: The um_convert utility (see TOOLS section) is a highly parallelized program that can convert the Abaqus input file to the MCNPUM file type. This file type is highly recommended when a complex geometry will be used more than once, as this will allow for faster problem setup times.

2.5.3 Ability to select overlap model by part:

One of the requirements for the UM implementation in MCNP was to permit multiple, non-contiguous, meshed parts instead of requiring one contiguous mesh. This leads to the possibility of overlapping parts. The code can accommodate a small amount of overlap in one of several ways using overlap models. The three overlap models currently in place in MCNP are EXIT, ENTRY and AVERAGE. The EXIT model, meaning that in an overlap situation the exit point of the overlap is used and results are accumulated accordingly. The second overlap model, ENTRY, is the one that uses the entry point of the overlap in an overlap situation and the results are accumulated accordingly. If the ENTRY point is behind the particle's current position, the current position is used; the particle never moves backwards. The third and last overlap model is called AVERAGE and results in averaging the entry and exit points in an attempt to find the midpoint of the overlap in the direction the particle is tracking; the particle's path length in the overlap is then divided between the two parts instead of being assigned to one or the other.

Although the code defaults to the EXIT model, ultimately the choice of which model to use is left to the user via the OVERLAP keyword on the EMBED card. If both parts are important and the particle flux through this region is fairly isotropic, the AVERAGE model is probably the best choice. If the flux is somewhat more directional and one part is deemed more significant than the other, a better choice might be ENTRY or EXIT; the user must decide. The user also has the ability to select the model to use by the instance/part (i.e., pseudo-cell) with the decision based upon the current instance/part in which the particle resides. For example, if the particle is currently in a part that specifies the EXIT model and the part into which it will travel specifies the ENTRY model, the EXIT model is used.

Note that extensive testing has been performed with the EXIT model but not with ENTRY and AVERAGE.

2.5.4 Ability to specify flux multipliers on the UM edits:

New to this version of MCNP is the ability to specify flux multipliers much like the existing capability of the FM card. This is allowed via the EMBEE keywords MYTPE = flux and FACTOR = <multiplicative constant>. Like the FMESH tallies, the UM edits cannot use the FM card attenuator sets.

2.5.5 Ability to read and handle multiple UM's in separate mesh universes:

For this version of MCNP a new capability for use with the UM feature now allows the user to request that the code read any number of Abaqus or MCNPUM files for use in separate mesh universes. No new input cards or parameters are required and the number of mesh files used is limited only by computer resources available.

2.6 CODE ENHANCEMENTS

Though there have been over a dozen code enhancements since the last release, this section provides a brief description of those that may be of most interest to users. A complete list of enhancements can be found in the Appendix.

2.6.1 Filenames used by MCNP version 6.2 may be up to 256 characters in length:

This extension will allow for more descriptive file naming conventions.

2.6.2 Permit line lengths up to 128 characters in MCNP input files and xsdir files:

For the past 40 years, all of the input for MCNP was limited to lines with a length of 80 characters. This limitation has been removed, and MCNP version 6.2 permits input lines of up to 128 characters. While this improvement seems trivial, the longer line length permits more spacing of input fields, reduces the number of continuation lines, and can greatly improve the clarity of input files.

2.6.3 Extend command line length to permit up to 4096 characters:

The command-line input may be up to 4096 characters to allow for more user flexibility and descriptiveness.

2.6.4 Removal of hard limit on bounding-surface-list entries for cell descriptions:

In defining cells (regions) in the MCNP input, part of the input is a list of bounding surfaces, with + or – to indicate sense and possible parentheses and union operators. In very old versions of MCNP, the length of the surface list (including operators) was limited to 999 entries. Over the years, this limit was raised occasionally, and is 9,999 in MCNP5, MCNP6.1, and MCNP6.1.1. Nevertheless, users continue to develop ever complex geometries where larger limits were needed. In MCNP version 6.2, this limit was entirely eliminated. MCNP version 6.2 examines the problem specifications and dynamically determines the space required for handling the boundary-list information.

2.6.5 The number of point detectors allowed increased from 100 to 1000:

Users now have a wider range of point detectors allowed for analysis.

2.7 VERIFICATION AND VALIDATION TESTS

Several of the Verification and Validation test sets have improved to allow users flexibility to run and view the test results.

2.7.1 VALIDATION_CRITICALITY AND VALIDATION_CRIT_EXPANDED

Perl scripts *run_val-crit.pl & run_val-critx.pl* have been added so that any system that has perl installed can run the validation tests VALIDATION_CRITICALITY and VALIDATION_CRIT_EXPANDED.

2.7.2 VALIDATION_SHIELDING

For VALIDATION_SHIELDING, the perl script *htm_results.pl* was added to the directory, which creates a set of web pages that plots both the test and the experimental results. The top web page is named *ValShld.htm*.

2.7.3 VERIFICATION_KEFF

The suite of analytic criticality benchmarks [21, 22] can now be run using the continuous-energy physics treatment, whereas previously these benchmarks could only be run using multigroup physics.

2.7.4 VERIFICATION_SHLD_SVDM

Added test results for ENDF/B-VII and ENDF/-VII.1 data sets from both MCNP5 1.60 and MCNP 6.2 runs. The perl script *htm_results.pl* was added to the directory, which creates a set of web pages that plots both the test and the experimental results. The top web page is named *VerShldSVDM.htm*.

2.8 TOOLS

With this release MCNP version 6.2 users will now have tools for criticality safety analysis, accessing MCNP output files, constructing radioactive source descriptions, as well as new and improved unstructured mesh utilities. The following sub-section provided brief descriptions of these tools along with references for further information.

2.8.1 Whisper – software for sensitivity-uncertainty-based nuclear criticality safety validation:

The Whisper-1.1 package [23] is available, including the Whisper code, supporting scripts, 1101 ICSBEP [24] benchmark problems (with input files and sensitivity profiles), and ACE-formatted covariance files for the nuclear data. Over 50 documents related to Whisper-1.1 are included in the MCNP Reference Collection [25]. The Whisper package provides sensitivity-uncertainty tools that may be used to support Nuclear Criticality Safety (NCS) validation.

Whisper is computational software designed to assist the NCS analyst with validation studies with the Monte Carlo radiation transport package MCNP. Standard approaches to validation rely on the selection of benchmarks based upon expert judgment. Whisper uses sensitivity/uncertainty (S/U) methods to select relevant benchmarks to a particular application or Area Of Applicability (AOA), or set of applications being analyzed. Using these benchmarks, Whisper computes a calculational margin from an extreme value distribution. In NCS, a Margin Of Subcriticality (MOS) that accounts for unknowns about the analysis. Typically, this MOS is some prescribed number by institutional requirements and/or derived from expert judgment, encompassing many aspects of criticality safety. Whisper will attempt to quantify the margin from two sources of potential unknowns, errors in the software and uncertainties in nuclear data. The Whisper-derived calculational margin and MOS may be used to set a baseline upper subcritical limit (USL) for a particular AOA, and additional margin may be applied by the NCS analyst as appropriate to ensure subcriticality for a specific application in the AOA.

Whisper provides a benchmark library containing over 1,100 MCNP input files spanning a large set of fissionable isotopes, forms (metal, oxide, solution), geometries, spectral characteristics, etc. Along with the benchmark library are scripts that may be used to add new benchmarks to the set. If the user desires, Whisper may analyze benchmarks using a generalized linear least squares (GLLS) fitting based on nuclear data covariances and identify those of lower quality. These may, at the discretion of the NCS analyst and their institution, be excluded from the validation to prevent contamination of potentially low quality data. Whisper provides a set of recommended benchmarks to be optionally excluded.

Whisper also provides two sets of 44-group covariance data. The first set is the same data that is distributed with SCALE 6.1 in a format that Whisper can parse. The second set is an adjusted nuclear data library based upon a GLLS fitting of the benchmarks following rejection. Whisper uses the latter to quantify the effect of nuclear data uncertainties within the MOS. Whisper also has the option to perform a nuclear covariance data adjustment to produce a custom adjusted covariance library for a different set of benchmarks.

Whisper is maintained as part of the MCNP6 Monte Carlo code distribution and adheres to the same rigorous SQA procedures.

2.8.2 The MCNPTools Package:

MCNPTools (version 3.8.0) provides access to output files MCNP produces, namely MCTAL, MESHTAL, and PTRAC files. MCNPTools is written in C++ and bound to Python. The source is distributed with the MCNP version 6.2 release so that users can build it. Precompiled C++ Linux (gcc 5.3.0), OS X (Apple Clang 8.0.0), and Windows (MSVC 19.00.24210) libraries are also distributed with the MCNP version 6.2 release. Additionally, prebuilt Python 3.6 packages that can be installed with the pip utility are also distributed with the MCNP version 6.2 release. Interested users should refer to the MCNPTools documentation provided with the release [26].

In addition to providing access to certain MCNP output files, MCNPTools comes with the following binary utilities to facilitate common tasks or query MCNP output files:

mergemctals – a utility to statistically merge MCTAL files. This utility is an experimental replacement for the merge_mctals Perl script provided in previous releases of MCNP.

mergemeshtals – a utility to statistically merge MESHTAL files. This utility is an experimental replacement for the merge_meshtals Perl script provided in previous releases of MCNP.

mctal2rad – a utility to produce TIFF formatted images from image tallies on MCTAL files.

l3dinfo – a utility to print information about LNK3DNT formatted files.

l3dcoarsen – a utility to coarsen the dimensions of a LNK3DNT formatted file.

l3dscale – a utility to scale the dimensions of a LNK3DNT formatted file.

2.8.3 Intrinsic Source Constructor (ISC):

ISC (version 1.3.0) is a software library and associated data files to construct radioactive source descriptions given a set of material isotopics. ISC is written in C++ and bound to Python. The source is distributed with the MCNP version 6.2 release so that users can build it. Precompiled C++ Linux (gcc 5.3.0), OS X (Apple Clang 8.0.0), and Windows (MSVC 19.00.24210) libraries are distributed with the MCNP 6.2 release. Additionally, prebuilt Python 3.6 packages that can be installed with the pip utility are also distributed with the MCNP version 6.2 release. Interested users should refer to the ISC documentation provided with the release [27].

In addition to being a library for generating source descriptions, ISC comes with the following binary utilities:

misc – the MCNP Intrinsic Source Constructor. This utility builds SDEF cards for radioactive source materials.

mattool – a utility to expand materials with "natural" ZAs into isotopic ZAs. Note that the isotopic ZAs produced may not have available cross sections provided with MCNP, and the user might have to make substitutions.

2.8.4 The UM_CONVERT utility program:

The um_convert (unstructured mesh convertor) program [28] is a command-line utility program that takes the information in the Abaqus input file and processes it with the UM input processing routines from REGL to produce the internal data structures that MCNP needs. The data from these internal data structures are written to a new file type, MCNPUM that MCNP can quickly read before launching into calculations. With the MCNPUM file type the UM input processing start up penalty need not happen every time the UM geometry is required. This can save substantial time for large mesh geometries that are used repeatedly.

2.8.5 Improved UM_POST_OP functionality:

Minimum and maximum values are now reported for the pseudo-tally option when using the UM_POST_OP utility [28]. UM_POST_OP will also now allow the user to write error histograms to an output file for all of the edits in the eeout file for which errors were requested.

3 Performance

MCNP version 6.2 has the same or better performance than MCNP6.1.1, demonstrating 1.5 to 2 times faster than MCNP6.1 for Nuclear Criticality Safety (NCS) applications, and is slightly faster than previous versions of MCNP5. For NCS applications, significant improvements were made in the energy and cross-section treatment for neutron calculations, the parallel threading performance, and efficient treatment of the checking for the multitude of options available in MCNP [29].

4 Additional Information

4.1 GENERAL INFORMATION

General users and practitioners should be aware of the following items related to changes in MCNP version 6.2, relative to the previous versions MCNP6.1 and MCNP6.1.1:

- MCNP version 6.2 includes all of the standard features for criticality calculations that have been available for the past 15 years, along with new features for sensitivity-uncertainty based methods for criticality validation. Only a few minor bug-fixes or enhancements were made to MCNP version 6.2 compared with previous versions, and these were extensively verified.

- MCNP version 6.2 was thoroughly verified against previous versions. In very many cases, results from MCNP 6.2 will match exactly results from MCNP6.1 or MCNP6.1.1, and in some cases results may differ but agree within combined statistical uncertainties. All things considered, MCNP version 6.2 results are as reliable as or more reliable than any previous release of MCNP.

- An immediate benefit of using MCNP version 6.2 (rather than MCNP6.1) is that the new version is typically 1.5-2 times faster.

- Users should be aware of the few instances where ACE data files were corrected and new versions released. Calculations involving zirc-hydride should be checked to determine whether erroneous data (described above) were used. Coupled neutron-photon calculations should also be checked to determine whether they would be affected by the previous lack of photon production data for hydrogen (described above). Calculations involving SiO_2 at high temperatures should also be checked.

- The coding changes to MCNP version 6.2 physics are relatively insignificant. Corrections to the $S(\alpha,\beta)$ thermal scattering numerics are generally negligible relative to problem statistics (or, in rare cases, prevent aborts). Similarly, the changes to adjoint-weighting for computing kinetics parameters may result in small differences, generally negligible compared to problem statistics.

- The change to the MCNP version 6.2 geometry treatment to correctly handle coincident surfaces in problems with *universe/fill* features will produce different round-off in the geometry tracking. This will produce differences in results relative to previous versions, but those differences should be small relative to problem statistics, and are not a concern. Any large differences that arise are an indication of previous (undetected) errors in older versions of MCNP. If any such cases are found, users should not hesitate to contact the

MCNP developers for assistance in further diagnosing the differences.

- It is standard practice that only validated computer codes, data, and computer systems be used. In verifying and validating MCNP version 6.2, the practitioner should carefully consider and review the verification-validation work reported by the MCNP developers, as well as the updates to the ACE nuclear data libraries. Users are encouraged to install and test the new release of MCNP version 6.2, with a goal of adopting it as soon as practical. Note that the last version of MCNP5 was released in 2010, and MCNP6.1 was released in 2013. Due to resource limitations, versions of MCNP that are more than 5 years old are no longer supported by the MCNP Team at LANL

- Be sure to check the MCNP FAQ section of the MCNP web pages at http://mcnp.lanl.gov for any advice, corrections, hints, etc., that may help with installation or trouble-shooting.

4.2 IEEE EXECEPTIONS IN MCNP

In the modern Fortran standards, calls to the Fortran "STOP" statement require the compiler to print any internal exceptions that are triggered during execution of a binary. Older versions of Fortran compilers did not implement this standard requirement, but newer versions of the compilers do. In Fortran, the intrinsic IEEE_EXCEPTIONS module and associated machinery keeps an account of various floating point operations that lead to IEEE_EXCEPTIONS. Instances of these exceptions are IEEE_INVALID, IEEE_DIVIDE_BY_ZERO, IEEE_OVERFLOW, IEEE_INEXACT, and IEEE_UNDERFLOW.

In MCNP, execution termination by reaching the end of program, user interrupt (CTRL-C followed by 'q' or 'k' for "quit" and "kill", respectively), or execution trouble (typically indicated by "bad trouble" messages) are all triggered with a Fortran "STOP" statement. Consequently, any IEEE exceptions that were triggered will print information to the console. **Users should pay particular attention to the information regarding IEEE_INVALID, IEEE_DIVIDE_BY_ZERO, and IEEE_OVERFLOW resulting from the IEEE_EXCEPTIONS module and should consider results suspect if these exceptions trigger**.

IEEE_UNDERFLOW and IEEE_INEXACT warnings are not usually important. IEEE_UNDERFLOW errors indicate that floating point numbers cannot be represented in "normal" arithmetic representation. In the case of IEEE underflows, the released MCNP 6.2 binaries are compiled to use "denormal" representations of the floating point values, i.e., not "normal" representation with reduced precision, until the values become even too small to represent in this fashion and they are set to zero. IEEE_EXACT warnings indicate that the result of a desired mathematical operation cannot be represented exactly in the available precision bits and that rounding, underflow, or overflow has occurred. Nearly all operations cause this exception to trigger (and it appears that most compilers opt not to report errors associated with this exception).

5 Software Quality Assurance

MCNP versions 6.x are developed with defined software engineering processes. All code changes are maintained under strict version control from inception to release. Code changes progress to being stored on developer branches to being merged into the common development trunk and finally into the production ready development head, from which code releases are made.

Rigorous change control (reviews and testing) is applied before incorporation of changes into the common development trunk or the production-ready head.

At all levels of development, thorough testing is accomplished through automated scripts and reporting. On developer branches, testing is invoked manually, but must be completed as a prerequisite for merging into the development trunk. The development trunk itself is automatically tested on multiple operating systems with multiple compilers and various parallel execution modes whenever changes occur and no less often than nightly.

The 40+ MCNP test suites include over 1500 regression, verification and validation tests to convey confidence that the LANL installations are correct and robust and that code changes do not adversely affect results. New tests are routinely added for enhancements, new features and bug fixes. A set is provided with the distribution to enable other locations to perform due diligence on local installations. Each release (including this one) is accompanied with installation instructions (and script) plus comprehensive V&V documentation against which other installations can be compared and verified.

MCNP 6.2, Whisper-1.1, the MCNP Reference Collection and the MCNP data files are all maintained under strict SQA procedures. The MCNP development team uses TeamForge software for source code and document version control and for tracking code features as well as bug tracking and resolution. MCNP software versions, coding, data, and test problems are maintained using TeamForge software along with Git version control for source configuration management suite for tracking all new features, modifications, changes, and documents.

6 Distribution and Installation

MCNP version 6.2 and Whisper-1.1 are available exclusively through the Radiation Safety Information Computational Center (RSICC) (http://www-rsicc.ornl.gov), based at the Oak Ridge National Laboratory in Oak Ridge, Tennessee. The code package consist of 3 DVD's which contain executables for Mac OS X, Linux, and Windows, the MCNP version 6.2 User Manual, source coding (if applicable), utility programs and scripts, the MCNP Reference Collection, non-ACE data files, the Whisper-1.1 package for nuclear criticality safety, testing suites for numerous areas; and other miscellaneous files. The codes only run on 64-bit operating systems.

The MCNP version 6.2 package is installed using a script file (bash or Windows), thus avoiding the need for special privileges. New in MCNP version 6.2, a log file records each step in the installation. Separate log files record all steps in the installation and testing processes. These log files may be retained to provide a complete record of the code installation and testing.

Step-by-step instructions on how to install MCNP version 6.2 are provided in the distribution package.

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Appendix - Closed Bugs, Code Enhancements, and Known Issues

Contained in this Appendix is a summary of code enhancements (Table 1) and known bugs that have been resolved since the last release of MCNP6.1.1beta (Table 2) along with known issues that still need to be addressed (Table 3). Users should familiarize themselves with these to ensure that the problems they have and will run are not impacted.

Table 1 Code enhancements for MCNP version 6.2

Tracking Number	Category	Description
artf36585	Data	Update the delayed-gamma line emission data file (cindergl.dat)
artf38880	Data	Allow use of delayed-gamma biasing with line data
artf33489	Error Messages	Clarify warning message if a material card contains an ZAID that is outside the model physics range
artf34641	Error Messages	Create a fatal error if DE/DF cards are used that have zeros for the lower limits and use logarithmic
artf34790	Error Messages	MCNP does not check the number of MATCELL entries in the EMBED card
artf38181	Error Messages	The code does not check that the values of delayed particle biasing keywords on the ACT card are entered in pairs.
artf32222	Miscellaneous	Excessive memory used if unresolved resonances are turned off and no materials are specified in the input file
artf33172	Miscellaneous	Allow tabs in the TMESH input section
artf32861	Output	Update FT ROC option to support large integers and allow output of data via FILES card.
artf35557	Parallel	Enable tracking with thread parallel execution when using the Compton imaging tally option (FT COM).
artf37147	Physics	In single event electron transport, logarithmic interpolations were changed to linear interpolation. The interpolation methods can be controlled by DBCN(81) through DBCN(89)
artf37461	Tally	The number of point detectors allowed is increased from 100 to 1000
artf44325	Tally	Allow F1 Tallies to use the FT card PHL option to obtain multiplicities
artf37200	Transport	The code limited number of burnup materials
artf37745	Unstructured Mesh	Reduce memory for multi-particle energy deposition edits.

Table 2 Bug Fixes for MCNP version 6.2

Tracking Number	Category	Description
artf35731	Compiler	Gfortran compiled code crashes with a segmentation fault when running keff sensitivity problem.
artf40146	Compiler	MCNP will not compile with the ints_8byte CONFIG option
artf31027	Data	The list of secondary particle data in neutron libraries sometime erroneously lists individual particle data as 'not present'
artf29075	Error Messages	The DF 99 dose option should be a fatal error
artf31704	Error Messages	DELETE or use what's in the 611 release notes: Some problems show xsdir lines printed to screen during initialization
artf39574	Error Messages	MCNP6 did not check for duplicate ZAIDS on a material card
artf10665	Geometry	When using rotations in a fill command, coincident surfaces (after the rotation) are not handled correctly during tracking, sometimes causing incorrect results or lost particles.
artf1305	Geometry	MCNP6 does not check for lattice fill error as thorough as MCNP5
artf18109	Geometry	If the PWT card is positive, and the source weight is very large (around 1e20 and above), results may be incorrect.
artf27105	Geometry	Magnetic field tracking loses particles if the field direction is perpendicular to a plane
artf31019	Geometry	The ELL Macro body surface card, when using foci the notation (r > 0), created an incorrect surface
artf34457	Geometry	If half or more of the surfaces in the input file are identical surfaces, tally results could be incorrect
artf4556	Geometry	EXT card on cell line gives fatal error
artf7529	Geometry	Duplicate IMP entries on cell cards not detected
artf18748	Integer Overflow	Integer overflow in DXTRAN diagnostics output
artf29128	Integer Overflow	Integer overflow prevents printing of tfc information
artf30637	Integer Overflow	Fortran format overflows in mctal files and when plotting
artf34661	Integer Overflow	If number of neutron collisions exceed 2G, negative values are printed to the outp file
artf38613	LNK3DNT Mesh	The LNK3DNT RZ and RZT tracking can fail if a transform in applied via a "fill" entry with parentheses
artf40242	Miscellaneous	Floating point precision issue can cause unrealistic relative error results the TMESH tallies
artf44220	Miscellaneous	MCNP6 will not find the cross section file if the full pathname is used in an xsdir file

Table 2 Bug Fixes for MCNP version 6.2 (continued)

Tracking Number	Category	Description
artf29785	Model Physics	If using point detectors, secondary particles from muon interactions can cause the code to stop with a "bad trouble error."
artf31044	Model Physics	If the COILF keyword value on the PHYS:N card is between 0.001 and 2.001, then light ion recoil is not handled correctly
artf32933	Model Physics	Evaporation particles from CEM and INCL were not isotropic, but biased in the +z direction
artf34699	Model Physics	On the LCA card, ipreq=3 and iexisa >0, ipreq was not set to 1 as described in the manual.
artf36427	Model Physics	Exotic particles that could not be transported by the LAHET models were not properly passed to LAQGSM model
artf36699	Model Physics	In around 1 out of 1e9 particles, floating point precision limitations cause a segmentation fault when LAQGSM processes elastic collisions
artf39917	Model Physics	When using the LLNL Fission model (method 5 on the FMULT card), if no neutrons are produced in the fission, any photons produced in the same event are ignored.
artf40855	Model Physics	If the first collision option (LCA card keyword NOARG, the 8th entry) is less than 0, NTER can be set to 0 for elastic scatter, which in turn can cause a PAX out-of-bounds error in LOSE_PAX
artf44484	Model Physics	The CGM excitation energy is adjusted by the Q value, but should only include the incident neutron energy in CMS plus the neutron binding energy
artf31180	Output	Diagnostic prints for large DXTRAN tracks were are not suppressed if the second entry of the DD card is zero
artf31759	Output	The energies listed in the "summary of photons produced by neutron collisions" table are inconsistent
artf32856	Output	If using the VOL card with an unstructured mesh, the masses of the cells whose volumes were changed are zero in the output file
artf36565	Output	If a source or tally comment card is indented, or has a four digit number, the comments are not written to the outp file correctly
artf38087	Output	If the TFC bin is zero for a tally that has the FT ROC treatment, the ROC table (Table 163) fails to print
artf39850	Output	CPU time was calculated incorrectly when running with threads
artf40850	Output	Values of the 'misses' section of the DXTRAN diagnostics table were not aligned properly
artf28041	Output Format	NPS integer write format limited to 11 digits in the mctal file
artf28704	Output Format	The weight window mesh keyword KMESH does not accept degrees or radians
artf38682	Output Format	Integers in the tally density plots overflow the write statement format
artf39553	Output Format	The format statement that writes the number of tally bin indices is limited to 4 digits

Table 2 Bug Fixes for MCNP version 6.2 (continued)

Tracking Number	Category	Description
artf31263	Parallel	MPI rendezvous not set correctly for FT ROC tally treatment
artf35551	Parallel	Spontaneous-decay sources give difference results if run with threading
artf38347	Parallel	Running with MPI with the SWAPB keyword on the BURN card produces incorrect results.
artf38905	Parallel	Some delayed-particle problems gave different results if run using MPI
artf39036	Parallel	The MVCALL array in LAQGSM was not reset to zero after each history, causing unintended results
artf39829	Parallel	If running with more than one thread, the results in Print Table 126 (the particle activity in each cell) are incorrect
artf26783	Physics	Time structure of delayed particles contains anomalous structure
artf28122	Physics	If the photon source energy is set to 100 GeV, the code exits with 'bad trouble' if the EPRDATA is used
artf28857	Physics	The density-effect-correction for electron transport is calculated incorrectly for mixtures
artf28953	Physics	Instead of what's stated in the User's Guide, if the value of the COND keyword a material (M) card is positive, the material is always set to a conductor
artf29966	Physics	MCNP6 can crash with a segmentation fault if the FMULT card is used and neutron energies are above 20 MeV
artf30139	Physics	A bad trouble error (non-existent law selected) can occur if charged particles are on the mode code, the neutron high energy cutoff is lowered to less than 16 MeV, and a low-energy neutron is transported.
artf32777	Physics	Code can hang, crash, or give wrong answers when the THRESH value on the ACT card is between 0.95 (default) and 1.0 (max).
artf38853	Physics	Charged particles using tabular data always use implicit capture, even if the weight cutoffs are set to analog capture.
artf47548	Physics	Values set on the 'gcut' keyword of the FIELD were ignored
artf28599	Plotter	Cross Section Plotter Fails to Ignore Coherent Scattering in Some Cases
artf28819	Plotter	Flux Image Tallies - FIC, FIP, and FIR plotted from the mctal file are not correct and can even cause the code to crash.
artf29279	Plotter	If plotting weight windows, and the WWP card (incorrectly) does not have a particle designation, MCNP crashes
artf31185	Plotter	Geometry Plotter can put cell labels in wrong place
artf33523a	Plotter	Fix format for printing 3-digit reaction numbers when using the command "xs ?" in the plotter
artf33523b	Plotter	Fix plotting of weight window values by color

Table 2 Bug Fixes for MCNP version 6.2 (continued)

Tracking Number	Category	Description
artf34284	Plotter	Fix 'color by cell' and 'color by den' options in the plotter
artf34665	Plotter	The boundary lines separating spherical mesh zones do not appear when plotting the mesh overlaid onto the geometry.
artf39377	Plotter	Long runtpe file name (>35) causes the fmesh plotter to crash
artf39573	Plotter	Proton libraries report incorrect MT values in cross section plotter
artf41534	Plotter	NPS integer overflow in tally plotting
artf42558	Plotter	The 'pause' plotting command with an argument causes the plotter to crash
artf47653	Plotter	Using mouse clicks with the interactive plotter prints the wrong value of the lattice index (ijk) for large lattices.
artf48162	Plotter	MCNP does not always create TMESH plots if the TMESH data is read from the mctal file
artf49245	Plotter	When plotting a FMESH tally from a runtpe file, the code crashes if an embedded mesh is included in the problem.
artf36673	Ptrac	For neutron analog capture, the ptrac file reports the event type as scatter, even though it could be absorption or fission
artf42556	Ptrac	Using a ptrac card with no keywords causes a segmentation fault
artf23313	Source	Incorrect source biasing if using the '-31' special function on the SB card
artf28646	Source	When using a source distribution that uses the A option such that the SP entries are the probability density, the particle weight drops to extremely low values without weight cutoff killing them
artf29404	Source	Print statement for low source cell efficiency can cause segmentation fault
artf29415	Source	Cosmic source option does not sample azimuthal angle correctly
artf29431	Source	Zero weight source particles can be created when using the Lal cosmic-ray spectrum model
artf29933	Source	If sampling for a cell in an explicitly defined lattice, the source routine can get into an infinite loop
artf34684	Source	MCNP6.1.1beta can hang if neutron tallies occur in void regions
artf35982	Source	The University of Delaware cosmic-source routine truncates the rigidities at latitudes >70 deg
artf37162	Source	The vertical input format (#) for SI and SP cards with cell entries fail if SI includes lattice indices
artf41006	Source	Round-off in LOC entries can result in compiler-dependent results for the choice of a cosmic ray background source

Table 2 Bug Fixes for MCNP version 6.2 (continued)

Tracking Number	Category	Description
artf42536	Source	On the SDEF card, need to allow the use of the NRM keyword with cosmic ray sources
artf44824	Source	RAD and EXT source distributions dependent on POS can only have 1 level of dependent source recursion
artf49187	Source	If using the Clem model for the cosmic source, the SDEF keyword NRM is ignored
artf21940	Tally	DXTRAN diagnostic table incorrect for kcode problems
artf27097	Tally	The TMC special tally option can give incorrect results if STEP is large (~e8 shakes), the path length in the cell is short (< 1 cm), or the time bin sizes are small
artf28324	Tally	If using the EPRDATA, photon fluorescence energy deposition was assigned to electrons, causing incorrect results for both photon and electron heating tallies
artf28387	Tally	Light ions tags set incorrectly, resulting in incorrect tag binning of energy deposition
artf29268	Tally	When transporting both electrons and photons using the EPRDATA, the energy deposition of fluorescence photons is counted twice.
artf29305	Tally	Enable specially tally option ROC with the PHL tally option
artf30048	Tally	Electron F6 tallies that use segmenting surfaces have incorrect results
artf31799	Tally	If using the DF card with IC=99 and FAC=-3, tally results are incorrect
artf31853	Tally	Nested DXTRAN spheres with lattices exit with 'bad trouble' error
artf32221	Tally	If using the PHL special tally treatment, and the TDEP keyword is used followed immediately by another specially tally option, default values of TDEP keyword are not set
artf32515	Tally	Use of the HPG-1 detector response on the FT PHL option results in all zeros for F6 and F8 tallies.
artf32541	Tally	Use of cyclic time bins can produce erroneous negative results.
artf33517	Tally	If a ZAID that used model cross sections is listed before a ZAID that uses data cross sections, the results of F6 tallies could be incorrect.
artf35403	Tally	If two ZAIDS are entered for the RES special tally option, MCNP creates a range of bins instead of just two bins, one for each ZAID.
artf37183	Tally	Point and ring detector results incorrect in multi-universe problems if the SUR keyword is used on the SDEF card
artf37560	Tally	FMESH tallies, when used with an FM card, do not process the photonuclear and proton cross sections correctly, causing incorrect results
artf37684	Tally	If both CAP and SCX are specified on the FT card, results are incorrect.

Table 2 Bug Fixes for MCNP version 6.2 (continued)

Tracking Number	Category	Description
artf37970	Tally	Contrary to the User's Guide, MCNP is not adjusting tally energy and time bins upper boundaries based on energy and time limits set in the problem.
artf38510	Tally	Incorrect contributions to photon point detector from particles inside the exclusion radius when using special tally treatment PDS option 1 or option 2
artf39260	Tally	If the PHYS:E keywords BNUM is > 1, or ENUM < 1, electron heating tallies results are incorrect
artf39871	Tally	The 2nd-order density perturbation terms are incorrect if more than one isotope is in the perturbed material
artf40868	Tally	The warning message for perturbations with density changes that are too low prints the lower limit as 1e-6 instead of the correct value, 1e-5
artf42295	Tally	On the KSEN card, if the cell keyword contains more than one value, the results for all the cell except the first one are all zero.
artf48928	Tally	When using the PHL special tally treatment, if the tally specified on the TDEB keyword is in the list of tallies after the PHL keyword, the tally used to trigger the PHL tallies might not be initialized.
artf30700	Transport	Incorrect creation time of secondary particles from electrons in magnetic field cells
artf31254	Transport	Curves surfaces of a LNK3DNT mesh can cause tracking errors
artf33526	Transport	First estimate of keff in adjoint calculations includes information from inactive cycles
artf35060	Transport	Burn-up material volumes are set incorrectly if the same material is used in cells with different volumes.
artf40885	Transport	Secondary particles that are killed because their energies are above the energy limit, were not recorded in the particle creation/loss table
artf49549	Transport	Charged particles except for electrons and positrons) do not track correctly in repeated structures.
artf31043	Unstructured Mesh	F8 tallies yield incorrect results with unstructured me
artf31200	Unstructured Mesh	Energy deposition embee edits produce NANs for void cells
artf31834	Unstructured Mesh	Electron transport in a unstructured mesh problem can put the code in an infinite loop
artf32640	Unstructured Mesh	If the name of a part in an Abaqus input file includes one of the keywords used to name an elset, unknown things can happen
artf32669	Unstructured Mesh	Eeout file contains multiple instances of the same edit.
artf32688	Unstructured Mesh	Edit results not attributed to correct element.
artf32748	Unstructured Mesh	Source cell rejection doesn't work correctly with unstructured meshes

Table 2 Bug Fixes for MCNP version 6.2 (continued)

Tracking Number	Category	Description
artf32955	Unstructured Mesh	Floating overflow error with 2nd order tetrahedron.
artf33010	Unstructured Mesh	Array out of bounds error on input processing.
artf33010	Unstructured Mesh	Fatal errors and segmentation faults on input processing OVERLAP keyword on embed card.
artf33416	Unstructured Mesh	Loosing particles with DXTRAN.
artf33417	Unstructured Mesh	Weight window banking of particles not banking correct cell number.
artf34047	Unstructured Mesh	Incorrect sampling from multiple UM volume sources.
artf34103	Unstructured Mesh	Incorrect cell number for source location.
artf34299	Unstructured Mesh	Non-consecutive element number in Abaqus input file.
artf34652, artf33556	Unstructured Mesh	Particles lost between CSG and UM systems.
artf34982	Unstructured Mesh	Continue run crashes when more particles specified on the mode card than there are corresponding embee cards.
artf35716	Unstructured Mesh	Energy deposition edits for charged particles not working.
artf36066	Unstructured Mesh	Attempt to fetch from allocatable variable NUM_COMPOS when it is not allocated.
artf36226	Unstructured Mesh	Incorrect material information put into the gmv file.
artf36228	Unstructured Mesh	Not finding correct cell with DXTRAN.
artf36748	Unstructured Mesh	Blank instance names in eeout file because of incomplete info in Abaqus input file.
artf36990	Unstructured Mesh	Code stalls with point detectors because of very small distances tracked along an edge shared by several tetrahedrons.
artf37031	Unstructured Mesh	Code crashes when trying to parse Abaqus input with incorrectly named material elsets – non-integers at the end of the elset name.
artf37166	Unstructured Mesh	Wrong cell number.
artf37247	Unstructured Mesh	Problem when running with mismatched eeout and input files.
artf37460, artf37363	Unstructured Mesh	Array out of bounds errors with point detectors / point detector tracking problems.
artf37471	Unstructured Mesh	Problems with 2nd order tetrahedron stalls.
artf37485	Unstructured Mesh	Speed issue with gamma problems. Banking check routine called when not needed.

Table 2 Bug Fixes for MCNP version 6.2 (continued)

Tracking Number	Category	Description
artf37916	Unstructured Mesh	Bank fill issue with secondary particles in overlap region.
artf37979	Unstructured Mesh	Incorrect edit setup resulting in bad eeout file.
artf38122	Unstructured Mesh	Edit information in eeout file for n,p composite edits was not correct for continue runs.
artf38152, artf37421	Unstructured Mesh	Uninitialized variables causing warnings about de-normalized numbers, etc.
artf38628	Unstructured Mesh	Not skipping over misplaced input in the assembly block of the Abaqus input file.
artf38750	Unstructured Mesh	Expand composite energy deposition edits to include particles other than neutrons and photons.
artf38833	Unstructured Mesh	Lost particles in proton runs.
artf39017	Unstructured Mesh	Fix format statement when writing out the number of elements for structured meshes. Restrict to 999999 or less.
artf39018	Unstructured Mesh	Disallow certain keywords on embed card for continue runs.
artf39019, artf38711.	Unstructured Mesh	Fix logic in tracking routines to allow use of both structured and unstructured mesh in the same problem.
artf39080	Unstructured Mesh	Electron & charged particle problems getting incorrect edit results.
artf39133	Unstructured Mesh	Producing secondary particles in overlap regions where one of the cells has zero importance.
artf42555	Unstructured Mesh	MPI slaves writing to fort.32 file.
artf43346, artf37486	Unstructured Mesh	Sampling source particles in an overlap region selected the incorrect cell.
artf44433	Unstructured Mesh	Lost particles when using overlap ENTRY model.
artf49225	Unstructured Mesh	If a problem contains multiple unstructured meshes and embee cards, MCNP crashes when attempting a continue run
artf34113	Unstructured Mesh Utilities	Um_post_op using wrong instance limits.
artf34114	Unstructured Mesh Utilities	When up_pre_op was building a skeleton input deck from the Abaqus input file, the correct material numbers were not used.
artf34115	Unstructured Mesh Utilities	When element checking with um_pre_op, incorrect information was printed because of an unallocated array.
artf35930	Unstructured Mesh Utilities	When generating pseudo-tallies, um_post_op was losing the file name and printing zero results.
artf36072	Unstructured Mesh Utilities	Multi-mesh capability did not allow um_post_op to merge files correctly.

Table 2 Bug Fixes for MCNP version 6.2 (continued)

Tracking Number	Category	Description
artf37215	Unstructured Mesh Utilities	Um_pre_op not writing correct background cell number when building a skeleton input file.
artf37285	Unstructured Mesh Utilities	When up_pre_op was building a skeleton input deck from the Abaqus input file, the mesh description was not stored correctly.
artf37767	Unstructured Mesh Utilities	Fix um_post_op for composite edits.
artf37915, artf34248	Unstructured Mesh Utilities	Indexing not correct when writing vtk file in um_post_op.
artf40342, artf39360	Unstructured Mesh Utilities	Uninitialized memory in um_pre_op.
artf29205	Utilities	Event Log Analyzer bugs & minor fixes

Table 3 Known Issues in MCNP version 6.2

Tracker Number	Category	Description
artf26747	Data	The light-ion data tables can give wrong tally results. When using light-ion data tables, MCNP can access arrays out of the array boundaries. Users should exercise extreme caution when employing light-ion data tables.
artf32570	Geometry	Invalid geometry errors can occur with multiple TR surfaces (e.g., planes & cylinders)
artf38507	Geometry	For some complicated cell definitions, MCNP does not calculate the cell volume and surface areas correctly. This is known to occur when highly complicated cells are described using multiple unions of the intersections of multiple surfaces. Although, volumes are incorrectly computed, tracking is correct and unaffected. If users choose to use such highly complicated cell descriptions, it is encouraged that they verify the volume stochastically as described in section 3.3.1.1 of the manual.
artf34662	Integer Overflow	When more than 2G particles are bank, the number of banked particles is reported as negative. This is due to an integer overflow problem in the number of banked particles.
artf40840	Integer Overflow	Integer overflows in dxtran diagnostics output are possible for large numbers of contributions to DXTRAN spheres. Negative values should be considered erroneous.
artf28495	LNK3DNT mesh	Calculations with an embedded structured (LNK3DNT) mesh give poor answers with or without multi-material elements. Care should be employed by users if using LNK3DNT mesh files.
artf33875	Miscellaneous	HISTP files are no longer written correctly by MCNP. Moreover, HTAPE3X (no longer released with MCNP) cannot process these HTAPE files.
artf40885	Miscellaneous	The particle creation/loss tables sometimes don't balance. The exact cause of this known bug is unknown, but may be due to particles being created above the maximum energy of that particle.
artf26745	Model Physics	If using the INCL/ABLA physics model, on rare occasions, the tally results from MPI run does not agree with serial run results
artf31028	Model Physics	Above 3.5 GeV, MCNP always uses LAQGSM, even if the user requests a different model.
artf28168	Output	No user bins (FU card) are printed to the mctal file for radiography tallies.
artf26662	Plotter	Contour plots do not normalize by tally bin widths.
artf28257	Plotter	Tally plotting command 'PRINTAL' does not list mesh tallies.
artf35219	Plotter	TMESH plots are not displayed overlayed with the geometry when using COM files.
artf38015	Plotter	For cells that have been rotated, the plotter will sometimes display the wrong cell number.

Table 3 Known Issues in MCNP version 6.2 (continued)

Tracker Number	Category	Description
artf44535	Plotter	Cannot plot a LNK3DNT mesh without a WWG card in the input file. A user might want to generate a LNK3DNT file using the MESH card and not generate a weight window. Currently for that situation the mesh won't plot.
artf42731	Ptrac	Spontaneous decay source particles call ptrak and eventp out of order. These out-of-order calls cause the source events to appear as events in the previous history.
artf25744	Source	If the SP card of a dependent distribution is before the SI card of the same distribution, and both cards have a letter before the distribution values, MCNP produced a fatal error during input.
artf30138	Source	Source transformations are not correct if used with a source distribution. The transformation number on the source distribution appears to be interpreted as the number of the distribution as it appears in the input, not its assigned number.
artf44063	Source	In a multi-universe problem, if the value of CELL keyword on the SDEF card refers to a cell lowest level universe, the other SDEF keywords may be ignored.
artf24140	Tally	Parenthesis in FM cards for FMESH tallies are not processed correctly causing MCNP to crash.
artf32429	Tally	When surface tallies are used in a lattice geometry, the tally can score in the wrong surface bin for a given lattice index. Only incoming surface crossings can currently be tallied. Outgoing crossings must be tallied in the lattice element they enter.
artf33067	Tally	the LET special tally treatment produces wrong results for electrons, unless DBCN(18) is set to zero.
artf36635	Tally	Heating corrections for Light-Ion recoils to avoid double-counting are applied to the wrong particle type.
artf38132	Tally	The dgeb and dneb features require pair of weight and energy pair. Thus, there need to be an even number of entries for the key word. The number of entries check is incorrect.
artf39260	Tally	If BNUM is > 1 or ENUM < 1, collision and electron heating tallies results are incorrect. This is a result of the energies of rouletted particles not being subtracted from the energy deposited for the F6 tally.
artf40520	Tally	If a FMESH tally includes a TR card, the untranslated coordinate values are written to the meshtal file.
artf40875	Tally	KPERT results incorrect if the RXN keyword is set to -7 (Total nu).
artf25474	Transport	Particles in cells with magnetic fields can get lost at large distances (10 km) from the center of the geometry.
artf25475	Transport	Particles in lattices cells magnetic fields can get lost.
artf33538	Transport	Rerun of a lost particle in a KCODE problem that has unresolved resonances tracks differently that the original history.

Table 3 Known Issues in MCNP version 6.2 (continued)

Tracker Number	Category	Description
artf42958	Transport	When tracking in a cell with magnetic fields, charged particle delta-rays are not created on the exact path of the deflected particle. This bug manifests as lost particles because delta-rays can appear to be created in inconsistent cells with particle that created them. Users should consider use of delta-ray production with magnetic fields to be incompatible features.
artf34418	Unstructured Mesh	If the volumer keyword is used on a DS card, and the source volume is not defined in the Abaqus mesh input file (i.e., no elset is tagged with the keyword "source"), MCNP will start the particles at the origin.

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