September 21, 2015

To: J.N. McKamy, Manager, USE DOE Nuclear Criticality Safety Program (NCSP)

From: Fitz Trumble, Chair, US DOE NCSP Criticality Safety Support Group (CSSG)

Subject: CSSG Tasking 2014-02 Response

In Tasking 2014-02 the CSSG was requested to provide guidance on the validation of criticality safety codes when only limited critical or exponential experiment data are available.

The CSSG Task 2014-02 Team Members were:

- Fitz Trumble (Team Leader)
- Dave Heinrichs
- Tom McLaughlin
- Glenn Christenbury (DOE-CSCT)

The attached response provides additional guidance and data sources that can be used when determining if sufficient data exists to declare a computational system validated. This tasking response was reviewed by the entire CSSG and comments were addressed into the final version provided here as an attachment.

cc: CSSG Members
M. Dunn
L. Scott
G. Christenbury, DOE-CSCT

Attachment: Response to CSSG Tasking 2014-02

Approval: 9/21/2015

Jerry McKamy, NCSP Manager

RESPONSE TO CSSG TASKING 2014-02

VALIDATION WITH LIMITED BENCHMARK DATA

EXECUTIVE SUMMARY

In tasking 2014-02 (Appendix A), the Criticality Safety Support Group (CSSG) was directed to provide additional guidance to support the development of a validation for systems with limited benchmark data while remaining in compliance with ANSI/ANS-8.24-2007, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations.* This tasking response provides the requested guidance as well as additional sources of data that can be used to determine if there is sufficient experimental data to declare a system validated for a particular application. A subteam of the CSSG, supplemented by a U.S. Department of Energy Criticality Safety Coordinating Team (DOE CSCT) resource as listed below, developed the tasking response. This response has been reviewed by the entire CSSG, and their comments have been incorporated; therefore, it represents a consensus opinion of that body.

The CSSG Task 2014-02 Team Members were:

- Fitz Trumble (Team Leader)
- Dave Heinrichs
- Tom McLaughlin
- Glenn Christenbury, DOE CSCT

According to section 6.1.3 of ANSI/ANS-8.24-2007, “The determination of bias uncertainty should contain allowances for uncertainties in benchmark physical properties and measurement techniques; uncertainties due to limitations in the geometric, material or neutronic representations (cross sections) used in the calculational model; and statistical and convergence uncertainties.” A closely related consideration is the statement in section 6.4 of ANSI/ANS-8.24-2007, “A margin of subcriticality shall be applied that is sufficiently large to ensure that calculated conditions will actually be subcritical.” It should be specifically noted however, that this guidance does not change the basic premise that if there is no experimental data, a bias meeting the intent of ANSI/ANS-8.1-2014, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, and ANSI/ANS-8.24 is not possible. This guidance is intended to supplement whatever existing critical and exponential experiment data is available.

The goal of this guidance is to help the user define the appropriate margins based on uncertainties within the data being evaluated. In the case where evaluation of the data shows that the uncertainties and sensitivities of the data are large enough that application of the derived margin has an impact on the ability of the process to be operated safely and efficiently, only two options are available: a redesign of the process or obtaining additional experimental data to reduce those uncertainties.

1.0 INTRODUCTION

The evaluation of the criticality safety of a process can be determined by a number of different methods including direct comparison to experiment (the most preferred method), comparison to values provided in handbooks of critical or subcritical limits, or calculation by validated criticality safety codes. ANSI/ANS-8.1-2014 states, in Section 4.2.7, that where applicable data are available, subcritical limits shall be established on bases derived from experiments. This section goes further to state that in the absence of directly applicable experimental measurements, the limits may be derived from calculations made by a method shown, by comparison with experimental data, to be valid. Section 4.3 of this standard goes on to define the practices and requirements for evaluating the validity of the method and recognizes the applicability of ANSI/ANS-8.24-2007 to neutron transport methods. ANSI/ANS-8.24-2007 goes further to provide additional detail about processes and techniques for the validation of computer-based neutron calculational methods used in nuclear criticality safety analysis. ANSI/ANS-8.24-2007 provides additional information on the selection of benchmarks, establishment of bias, and margins and on extending the area of applicability of the validation beyond experimental data.

Where there is a multitude of experimental critical or exponential data points with which to perform a validation, guidance within the existing standards (ANSI/ANS-8.1 and 8.24) is sufficient and will not be further covered within this tasking response.

For those cases where there is no critical or exponential data, subcritical limits can still be determined; however, the system would not be "validated" within the scope of ANSI/ANS-8.24. Many of the values of subcritical limits generated within ANSI/ANS-8.15, *Nuclear Criticality Control of Special Actinide Elements*, fall into this category. It is noted that all the subcritical limits in ANSI/ANS-8.15, as well as the estimated critical masses therein, were generated based on a combination of very limited small-sample replacement measurements and sparse neutron cross section data. Thus these values were generated without being able to meet the requirements of either ANSI/ANS-8.1 or 8.24; however, as is noted in footnote 4 of Section 4.3, "use of subcritical limit data provided in ANSI/ANS standards or accepted reference publications does not require further validation."

Within the ANSI/ANS-8.1 and 8.24 standards there is, however, little information to guide the user concerning approaches that can be taken when there is only a limited amount of critical or exponential experimental data available.

The CSSG was tasked with proposing additional guidance for developing an appropriate validation in cases with limited quantity benchmark data. The guidance should address the determination of bias, bias uncertainty, and validation adequacy and include consideration of using other data in an acceptable validation approach.

2.0 CONSIDERATIONS WHEN EVALUATING SYSTEMS WITH LIMITED BENCHMARK DATA

2.1 DETERMINATION OF AN ADEQUATE NUMBER OF BENCHMARKS

The ANSI/ANS-8.24 standard does not define what constitutes a sufficient number of benchmarks for an adequate validation. This is because the number of benchmarks needed depends on their quality, similarity to the evaluated system and the methods used to determine bias and bias uncertainty. For a system that has a nearly exact, carefully performed and well documented experiment, evaluation of as little as one experiment may be sufficient to determine a subcritical limit. As the difference between the benchmark data and the system under evaluation grows, the expectation for additional experimental data to determine and quantify these differences (bias and uncertainty) also grows. If statistical methods are being employed to generate a bias, then, as noted in NUREG/CR-6698, *Guide for Validation of Nuclear Criticality Safety Calculational Methodology,* many statistical methods are not valid or are highly uncertain without a minimal set of data points (dependent on the statistical method used). Critical and exponential experimental data may be supplemented by other available information to determine if the collection of information creates adequate experimental information for validation. While a bias (and its uncertainty) cannot be established without any experimental data, it may be established with limited data. The use of such a bias in the establishment of a subcritical limit must be accompanied by a subcritical margin that gives adequate consideration to the type, quantity, and quality of the limited data.

2.2 USE OF DIFFERENTIAL CROSS SECTION INFORMATION FOR THE MATERIAL/CROSS SECTION OF INTEREST

Appendix B provides information on where to find and how to evaluate differential cross section information. This information can be used to directly estimate the uncertainty of a cross section of interest.

Given the uncertainty in a particular cross section of interest, direct perturbations can be used to determine the sensitivity of the reactivity of the system being evaluated to that uncertainty. The cross section uncertainty can also be coupled with sensitivity and uncertainty methods as described in the next section to provide estimates of the impacts of those uncertainties on the reactivity of the system.

Another approach would be to use the benchmark descriptions but with a cross section of interest from another nuclear data library (e.g., JEFF, JENDL) replacing the one of interest. This provides some information as to the potential uncertainty associated with the processing of the evaluated data and its potential effects on the reactivity of the system. Care should be taken to understand the independence of the data and the sources used in its generation (as noted in Appendix B).

Application of this uncertainty data could be used to increase the calculational uncertainty of the experiments used or more likely to justify that the subcritical margin selected is adequate to cover the cross section uncertainty effects (or to judge them negligible in relation to the other uncertainties evaluated).

2.3 USE OF SENSITIVITY STUDIES AND COVARIANCE DATA[1]

Sensitivity studies may be used to identify the nuclides that are unimportant to the validation, per guidance provided in this section, and for which limited data exist. If this is the case, the nuclide can be ignored and the computational method can be validated for the application using traditional methodologies. This is discussed further in Section 2.3.1 below.

If sensitivity studies demonstrate the nuclide of concern is important, the bias in the computational method cannot be determined without some experimental data.

For those situations where a nuclide is determined to be important and limited data exist, validation may still be possible. However, an additional margin should be used to compensate for the limited data. This margin is separate from, and in addition to, any margin needed for extending the benchmark applicability to the validation. Sensitivity and uncertainty tools may be used as part of the technical basis for determining the magnitude of the margin. This is discussed in Section 2.3.2 and 2.3.3 below.

2.3.1 Sensitivity studies may be used to justify omission of any unimportant application nuclide from the computational method validation. Whenever this approach is taken, justification for excluding the nuclide from the validation should be documented and the level of technical justification provided should be commensurate with the influence the omitted nuclide is estimated to have on the reactivity of the application (i.e., the greater the potential influence, the greater the burden of proof for its omission).

Technical justification for nuclide omission can be developed using various approaches, such as:

a. direct comparison of the application system nuclide’s macroscopic cross section, which demonstrates its reaction rate is insignificant when compared to the more dominant nuclides in the system;

b. simple direct perturbation calculations, which demonstrate that variations in the nuclide’s abundance will not result in a significant change in the application system reactivity; or

c. sophisticated computation methods demonstrating that the sensitivity of the application system to the nuclide being omitted is insignificant compared to other, more dominant nuclides (e.g., *Tools for Sensitivity and Uncertainty Assessment Methodology Implementation*, TSUNAMI, included with the SCALE code system, (ORNL/TM-2005/39, *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*) or as discussed in the Adjoints, Perturbations, Sensitivity Analysis section of the MCNP Reference Collection distributed with MCNP 6).

[1] The CSSG writing group would like to acknowledge Brad Reardon of Oak Ridge National Laboratory (ORNL) for contributions to section 2.3 of this tasking response dealing with the capabilities of SCALE and the use and data interpretation of TSUNAMI-IP. While SCALE is used often as an example in this report, MCNP and other mature codes also have this capability.

The exclusion of nuclides that do not have an appreciable effect on the reactivity of the system is consistent with logic laid out in U.S. Nuclear Regulatory Commission (NRC) Division of Fuel Cycle Safety and Safeguards Interim Staff Guidance 10 (FCSS-ISG-10), *Justification for Minimum Margin of Subcriticality for Safety*:

"… critical experiments should include any materials that can have an appreciable effect on the calculated k_{eff}, so that the effect due to the cross sections of those materials is included in the bias. Furthermore, these materials should have at least the same reactivity worth in the experiments (which may be evidenced by having similar number densities) as in the applications. Otherwise, the effect of any bias from the underlying cross sections or the assumed material composition may be masked in the applications. The materials must be present in a statistically significant number of experiments having similar neutron spectra to the application. Conversely, materials that do not have an appreciable effect on the bias may be neglected and would not have to be represented in the critical experiments."

2.3.2 Estimates of application system reactivity uncertainties, based on the use of nuclear parameter covariance data, are useful in assessing an additional safety margin to account for these materials, but may not be used to justify omission of application system nuclides from the computational method validation. This restriction stems from historical uncertainties associated with the covariance data. The process of generating a complete covariance library necessitates the use of nuclear physics models and codes, evaluator estimates, and expert judgment ("Sensitivity and Uncertainty Analysis Capabilities and Data in SCALE," *Nucl. Technol*). There are still reservations about the quality and utility of the current "state-of-the-art" covariance data ("ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields, and Decay Data," *Nucl. Data Sheets*), although recent attention to these deficiencies has stimulated improvements in covariance evaluations. For example, the ENDF/B-VII.1 nuclear data includes approximately 200 nuclides with covariance data ("Sensitivity and Uncertainty Based Criticality Safety Validation Techniques," *Nucl. Sci. Eng*.) and will be included in the SCALE 6.2 code package. Means to quantify, or validate, the reliability of the uncertainty data ultimately produced by these processes are emerging, with comparisons between observed biases and those predicted by uncertainty quantification documented in the SCALE 6.1 criticality safety validation report (ORNL/TM-2011/450, *Criticality Safety Validation of Scale 6.1*).

2.3.3 For those situations where a nuclide is determined to be important, only limited data exist, and validation of the computation method is asserted, an additional margin should be used to account for the limited data. Sensitivity and uncertainty tools[2] may be used as part of the technical basis for determining the magnitude of the margin by giving consideration to 1) the similarity between the experimental system(s) (i.e., the limited data) and the application and 2) the estimated uncertainties associated with the nuclear data for the nuclide(s) of concern.

[2] Correct use of these methods requires an understanding of the tool, its application, and often the use of qualitative judgment.

1. A smaller margin would be appropriate when the available experimental data are associated with systems more similar to the application. Conversely, a larger margin is appropriate when there is less similarity. One way to assess the similarity is by using the sensitivity and uncertainty capabilities in SCALE TSUNAMI-IP (i.e., the TSUNAMI-IP correlation coefficient, c_k, value). If this capability is used, only experimental configurations with sufficiently high correlation coefficients should be considered. A threshold value for the correlation coefficient depends on the application and should be carefully assessed. Past studies have indicated that systems with c_k values of 0.9 and above are highly similar to the application, those with values of 0.8 to 0.9 are marginally similar, and those with values <0.8 may not be similar in terms of computational bias ("Sensitivity and Uncertainty Based Criticality Safety Validation Techniques,").

   NRC guidance states the following in FCSS-ISG-10:

   "The NRC staff currently considers a correlation coefficient of c_k > 0.95 to be indicative of a very high degree of similarity. This is based on the staff's experience comparing the results from TSUNAMI to those from a more traditional screening criterion approach. The NRC staff also considers a correlation coefficient between 0.90 and 0.95 to be indicative of a high degree of similarity. However, owing to the amount of experience with TSUNAMI, in this range use of the code should be supplemented with other methods of evaluating critical experiment similarity. Conversely, a correlation coefficient less than 0.90 should not be used as a demonstration of a high or very high degree of critical experiment similarity."

2. In general, comparisons of uncertainties in k_{eff} due to nuclear data uncertainties (even at the one standard deviation level) to the computational bias seen in well represented experimental systems, over a broad range of experimental arrangements, have indicated that the uncertainty quantification provides bounding estimates of the observed computational bias (ORNL/TM-2011/450, p. 63). Accordingly, it is possible to bound the computational bias introduced by a particular nuclide, for which little experimental data are available, by examining the k_{eff} uncertainties introduced by the uncertainties in that nuclide's nuclear data. One way to do this is by using the sensitivity and uncertainty capabilities in SCALE TSUNAMI. The additional margin should be at least as large as the k_{eff} uncertainties introduced by the uncertainties in that nuclide's nuclear data (at the one sigma level). If more than one nuclide is involved, the margin should be as least as large as the square root of the sum of the squares of the uncertainties for all the nuclides involved.

   This approach is applied in NRC ISG-8-R3, *Interim Staff Guidance 8, Revision 3*, for the use of fission product burnup credit in pressurized water reactor spent fuel in transportation and storage casks. Once major contributors to k_{eff} are validated against experimental data, an additional margin is assigned to allow the inclusion of reactivity credit for neutron absorbing fission products for which integral experiment data for validation are not available. ISG-8-R3 reports:

> "The TSUNAMI code is used to propagate the cross section uncertainties represented by the covariance data into k_{eff} uncertainties for each fission product isotope used in a particular application. The theoretical basis of this validation technique is that computational biases are primarily caused by errors in the cross section data, which are quantified and bounded, with a 1σ confidence, by the cross section covariance data."
>
> "This methodology has been benchmarked against a large number of low enrichment uranium (LEU) critical experiments, high enrichment uranium (HEU) critical experiments, plutonium critical experiments, and mixed uranium and plutonium critical experiments to demonstrate that the k_{eff} uncertainty estimates generated by the method are consistent with the calculated biases for these systems. The k_{eff} uncertainty results for specific fission products were also compared to fission product bias estimates obtained from the limited number of critical experiments that include fission products."
>
> Additional details of the approach for providing reactivity credit for the absorber materials without experimental validation data are provided in NUREG/CR-7109.

2.4 EVALUATION OF THE BEHAVIOR OF THE CROSS SECTION BEING CREDITED

Most often the nuclide of interest is incorporated in either a structural component, such as the material of construction of a process vessel, is a soluble or insoluble neutron poison, or is an impurity-level element/nuclide in the fissile material. In almost all cases it is an absorbing, but non-fissioning, element/nuclide. If the evaluation of the maximum multiplication factor of the system for the bounding credible upset model indicates that the negative reactivity of the element/nuclide is important to maintaining the k_{eff} below the subcritical limit, then the analyst should research both the neutron energy spectrum in the region containing the nuclide of interest and then the microscopic cross section data (and its uncertainty) for this nuclide.

With this information at hand, adjustments can be made to the cross sections used in the application models to evaluate their effects. Simplistically this can sometimes be accomplished by adjusting the number density of the element/nuclide itself. If the nuclide is predominantly a thermal absorber in the calculational model and also exhibits a 1/v shape in the thermal range, then surrogate nuclides can be evaluated to separate specific effects. Nuclides that have this same absorption cross section shape but with differing resonance region to thermal absorption cross sections and different scatter-to-thermal absorption ratios may be utilized to separate out the thermal absorption from resonance absorption and scatter effects on k_{eff}. Use of such "replacement" analysis where a similar cross section (which has significant benchmarking) is used to replace one (which while well behaved may not have as much experimental data) can provide an estimate of the reactivity worth of the nuclide cross section under evaluation and determine either an estimate of the uncertainty (which can be factored into the benchmark uncertainty and would impact the bias uncertainty) or can be used to justify the selection of the subcritical margin to account for this effect.

2.5 EVALUATION OF AVAILABLE MARGIN IN THE SYSTEM BEING EVALUATED

An important consideration in determining the acceptable level of experimental data to perform validation applicability is the amount of margin available for the system being evaluated. For example, if the bounding credible upset condition being evaluated has a k_{eff} of 0.5, there is very little likelihood that an error in a cross section could represent a 100% increase in reactivity. A quick and easy method to determine the order of magnitude effect of such a cross section error would be to re-run the calculation without the material of interest (for those materials credited for absorption or scattering). If the difference in k_{eff}, due to removal of the material, is insignificant or is significantly less than the available margin then that result can be factored in during determination of the amount of acceptable data needed for validation.

This section provides some general rules of thumb on the level of rigor in the validation that would be expected based on the amount of available margin.

- For those cases where k_{eff} for a normal or credible upset condition is below 0.8 for the evaluated systems, validation can be general and can tolerate significant uncertainty in the cross sections and modeling assumptions.

- For systems where the k_{eff} is between 0.8 and 0.9 attention should be paid to having a more detailed understanding of the uncertainties and in selecting appropriate margins. Care should be taken to understand trends in validation data and to evaluate basic cross section uncertainties. If there are limited critical or exponential data available, considerations should be given to using differential data to evaluate uncertainty and sensitivity of those cross sections dominating the k_{eff} results.

- For those cases where the calculated k_{eff} is above 0.9, significant care should be taken in the selection of benchmarks, both critical and exponential, to ensure adequate effort has been taken to "eliminate to the extent possible" systemic biases or uncertainties by using multiple data sources for the validation. If limited critical or exponential data are available, and it is not clear that the experimental data available closely match the system being evaluated, then examination of differential data, cross section behavior and/or covariance data to estimate appropriate sensitivities and uncertainties in the system should be used.

2.6 SELECTION OF THE MARGIN OF SUBCRITICALITY FOR SYSTEMS WITH LIMITED VALIDATION DATA

For systems with limited validation data, the most straightforward way to account for the paucity of experimental data is to apply a margin sufficient to account for the uncertainties in the method and data. Evaluations of various data sources, as noted in the sections above, can provide a good estimation of the uncertainty and sensitivity of the system to those methods and data and provide information supporting the correct selection of the margin. If the application of that margin to the evaluated cases results in acceptable operation of the process, then the system should be considered to be adequately validated. In the case where evaluation of the data shows that the uncertainties and sensitivities of the data are large enough that application of the necessary margin has an impact on the ability of the process to be operated safely and efficiently, only two options are available: a redesign of the process or obtaining additional experimental data to reduce those uncertainties and the assigned margin.

3.0 CONCLUSION

Guidance for and examples of validations in ANSI/ANS-8.24 focus on systems with numerous experimental data and on the use of statistical methods. For systems with elements/nuclides for which there is little critical or subcritical data other approaches are required. Initially the analyst should evaluate the sensitivity of the system to the elements/nuclides of concern. If k_{eff} is insensitive to these nuclides, then the problem may be solved.

If the element or nuclide of concern has a significant delta-k contribution for the normal or bounding credible upset model then it may be necessary to investigate which microscopic cross section is the major contributor to the total effect, e.g., absorption or scatter. Once the major contributor to the system reactivity is determined then one can research the actual measurement uncertainty of the data. While this may seem laborious, if successful it will always be far less expensive and less time-consuming than generating new experimental critical mass data.

If it is determined that the paucity of data would require selection of margins that preclude safe and efficient operation, then the process will either need to be redesigned or additional data will need to be generated to reduce the uncertainties.

The CSSG conclusion is that when used to supplement critical or exponential data, the approaches discussed in this tasking response fall within the intent of the ANSI/ANS-8 series of standards on validation of calculational methods.

4.0 REFERENCES

"ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields, and Decay Data," *Nuclear Data Sheets*, Volume 112, Issue 12, pp. 2887-2996, M.B. Chadwick et al, December 2011.

"Sensitivity and Uncertainty Analysis Capabilities and Data in SCALE," *Nuclear Technology*, Volume 174, Number 2, pp. 236-288, B.T. Rearden, M.L. Williams, M.A. Jessee, D.E. Mueller, and D.A. Wiarda, May 2011.

"Sensitivity and Uncertainty Based Criticality Safety Validation Techniques," *Nuclear Science Engineering*, Volume 146, Number 3, pp. 340-366, B.L. Broadhead et al., March 2004.

ANSI/ANS-8.1-2014, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, American National Standards Institute/American Nuclear Society, La Grange Park, IL.

ANSI/ANS-8.24-2007, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*, American National Standards Institute/American Nuclear Society, La Grange Park, IL.

FCSS-ISG-10, Rev. 0, *Justification for Minimum Margin of Subcriticality for Safety*, Interim Staff Guidance for Fuel Cycle Facilities, U.S. Nuclear Regulatory Commission, Washington, DC.

NRC ISG-8-R3, Rev. 3, *Burnup Credit in the Criticality Safety Analysis of PWR Spent Fuel in Transportation and Storage Casks*, Interim Staff Guidance 8, U.S. Nuclear Regulatory Commission, Washington, DC.

NUREG/CR-6698, *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*, J.C. Dean et al., prepared for the U.S. Nuclear Regulatory Commission by SAIC, January 2001.

NUREG/CR-7109 (ORNL/TM-2011/514), *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Criticality (k_{eff}) Predictions*, J.M. Scaglione, D.E. Mueller, J.C. Wagner, and W.J. Marshall, prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, April 2012.

ORNL/TM-2005/39, Version 6.1, *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,* Oak Ridge National Laboratory, Oak Ridge, TN, June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785 (2011).

ORNL/TM-2011/450, *Criticality Safety Validation of Scale 6.1*, W.J. Marshall and B.T. Rearden, Oak Ridge National Laboratory, Oak Ridge, TN, November 2011.

APPENDIX A. CSSG TASKING 2014-02

CSSG TASKING 2014-02
Date Issued: June 4, 2014

Task Title: ***Validation with Limited Benchmark Data***

Task Statement:

ANSI/ANS-8.24 provides guidance on performing validations of computer codes for criticality safety analyses. However it provides little in the way of guidance if the benchmark data set for any specific element or isotope, to be specifically credited for some feature (e.g. neutron absorption) is of limited quantity (e.g., in the specific case of titanium there are two critical benchmarks). The CSSG is directed to evaluate and propose additional guidance for evaluating cases with limited quantity benchmark data into an appropriate validation.

The guidance provided should address the determination of the bias, bias uncertainty, and the validation adequacy, in compliance with the intent of ANSI/ANS-8.24. Guidance should also be provided on what additional data or considerations (e.g., consideration of differential data, evaluated data uncertainties) would be necessary for an acceptable validation approach?

Finally, the CSSG should consider populating the guidance with a specific example (e.g., titanium absorption in a thermal spectrum).

Resources:

CSSG Task 2014-02 Team Members:

- Fitz Trumble, CSSG Team Leader
- Dave Heinrichs, CSSG
- Tom McLaughlin, CSSG
- Glenn Christenbury, CSCT

Contractor CSSG members of the team will use their FY14 NCSP CSSG support funding as appropriate; DOE CSSG members of the team will utilize support from their site offices. It is up to the team members to utilize other expertise, or include other interested parties, as can be made available to support the tasking, without incurring additional CSSG expenses. No travel is anticipated to be necessary to support this tasking.

Task Deliverables:

1. CSSG Subgroup to hold task 'kickoff' telecom by June 16, 2014
2. CSSG Subgroup to provide draft guidance to full CSSG for review: July 30, 2014
3. Full CSSG to provide review of guidance to Task Team Leader: August 28, 2014
4. CSSG Subgroup to provide finalized guidance to NCSP Manager: September 30, 2014

Task Completion Date: September 30, 2014

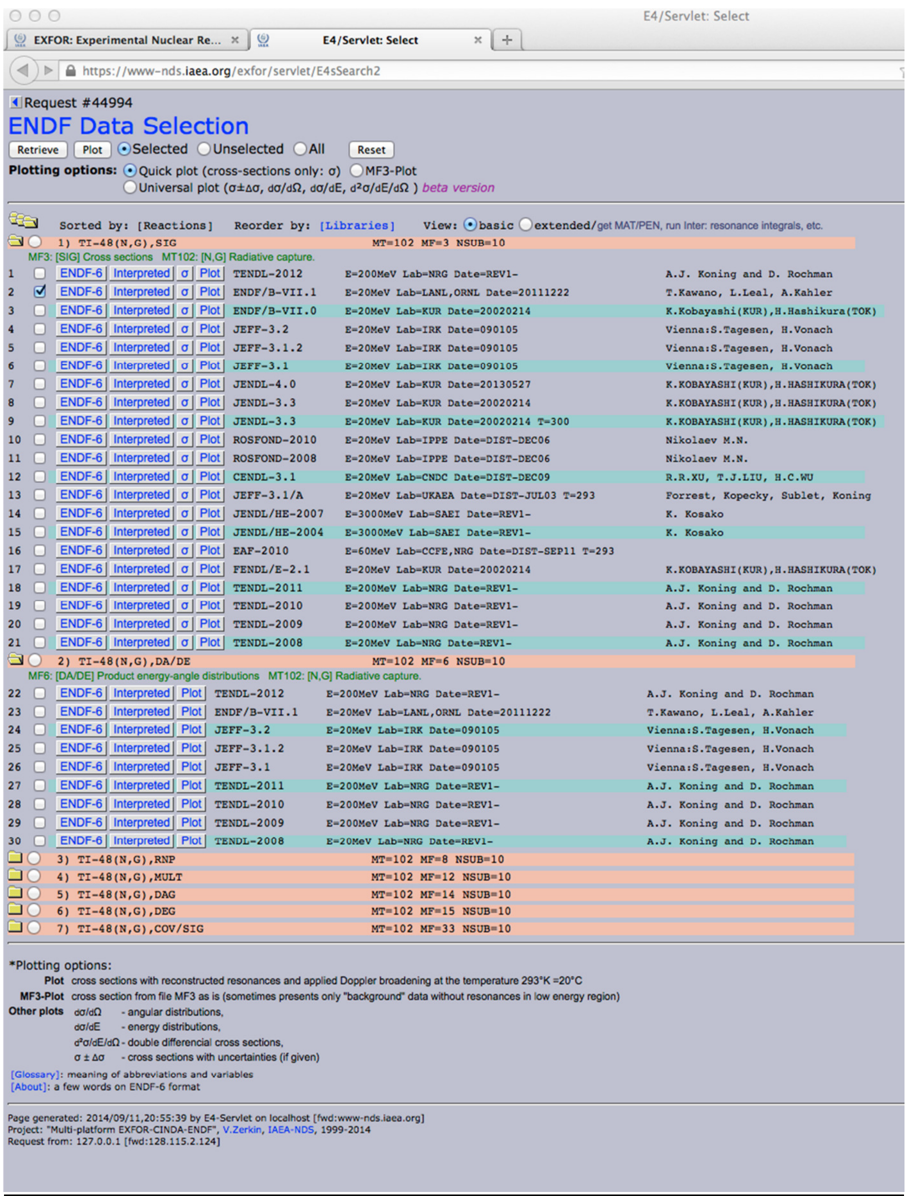
Signed: Jn McKamy
Jerry N. McKamy, Manager US DOE NCSP
Director NA-00-10

APPENDIX B. SOURCES AND EVALUATION OF DIFFERENTIAL CROSS SECTION DATA

Beyond just critical and exponential data, there exists significant information (data and uncertainties) available for differential data associated with various cross sections of materials important to the determination of a system's reactivity. Information on thermal cross sections and resonance integrals and their uncertainties is published by S.F. Mughabghab, "Atlas of Neutron Resonances," Elsevier Science (2006); see http://www.nndc.bnl.gov/atlas/.

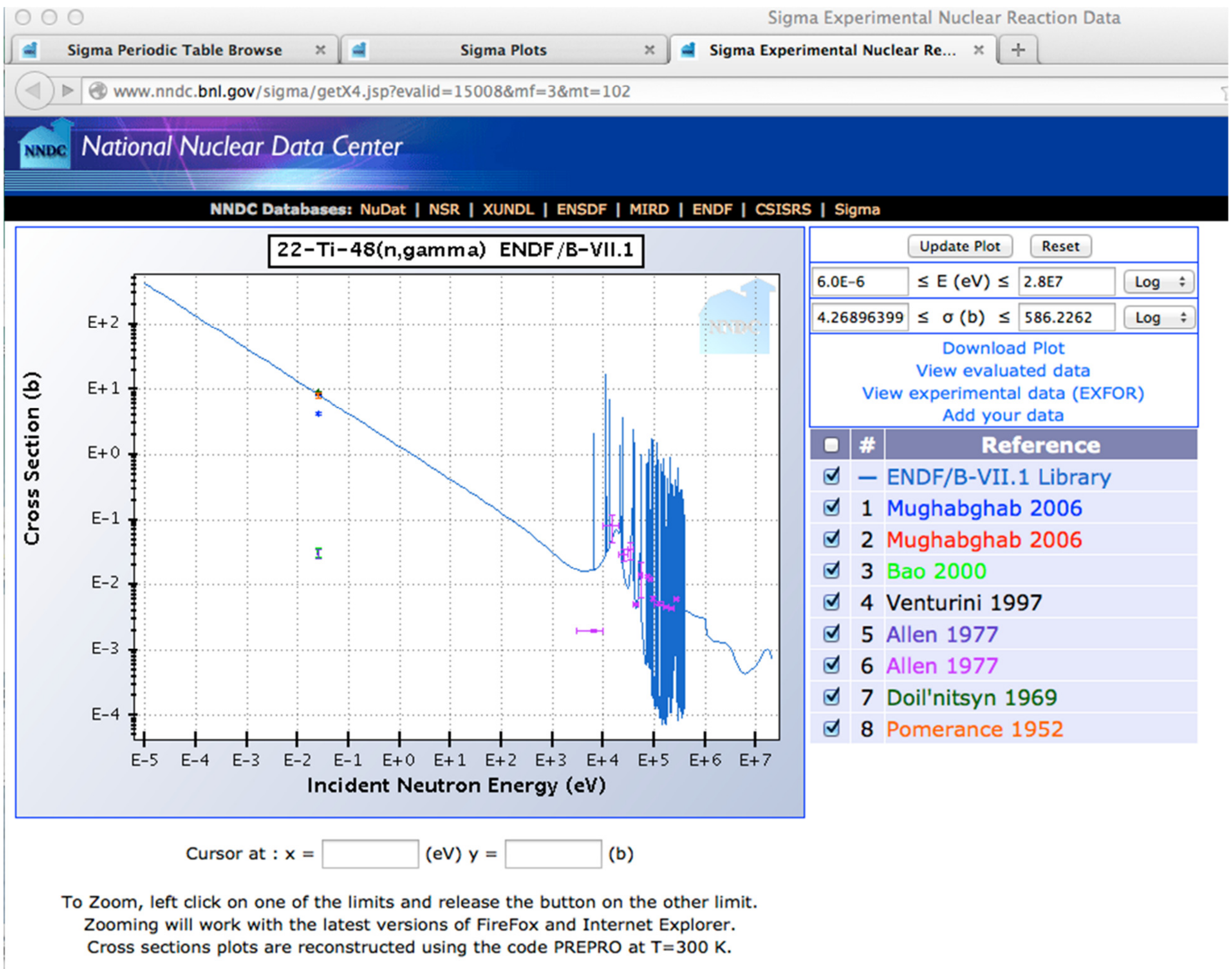
For individual reactions, the cross section data can be easily obtained either as text or as a graph using the online EXFOR GUI maintained by the International Atomic Energy Agency (IAEA). An example showing the GUI for Ti48(n,g) is given in Figure B-1.

Figure B-1. EXFOR GUI



In this case data from several evaluations is available including ENDF/B-VII.1 (USA), JENDL-4.0 (JAPAN) and JEFF-3.2 (Europe). While covariance data is available, it is often very useful to plot the evaluated data together with experimental data showing experimental uncertainties. This can be done online using the SIGMA Evaluated Nuclear Data File (ENDF) Retrieval & Plotting GUI maintained by the National Nuclear Data Center (see http://www.nndc.bnl.gov/sigma/index.jsp?as=48&lib=endfb7.1&nsub=10). As an example, data for Ti48(n,g) is shown in Figure B-2. By clicking on "View experimental data (EXFOR)", the user can also obtain the experimental cross section data and its uncertainty.

Figure B-2. SIGMA ENDF GUI



Another resource is the Computer Index of Nuclear Reaction Data (CINDA), which includes a GUI maintained by the IAEA (see https://www-nds.iaea.org/exfor/cinda.htm) that may be used to obtain citations to the published literature for a particular reaction. An example for thermal neutron radiative capture (n,g) for Ti-48 is provided here. The GUI is illustrated in Figure B-3 with search results are shown in Figure B-4.

Figure B-3. CINDA GUI

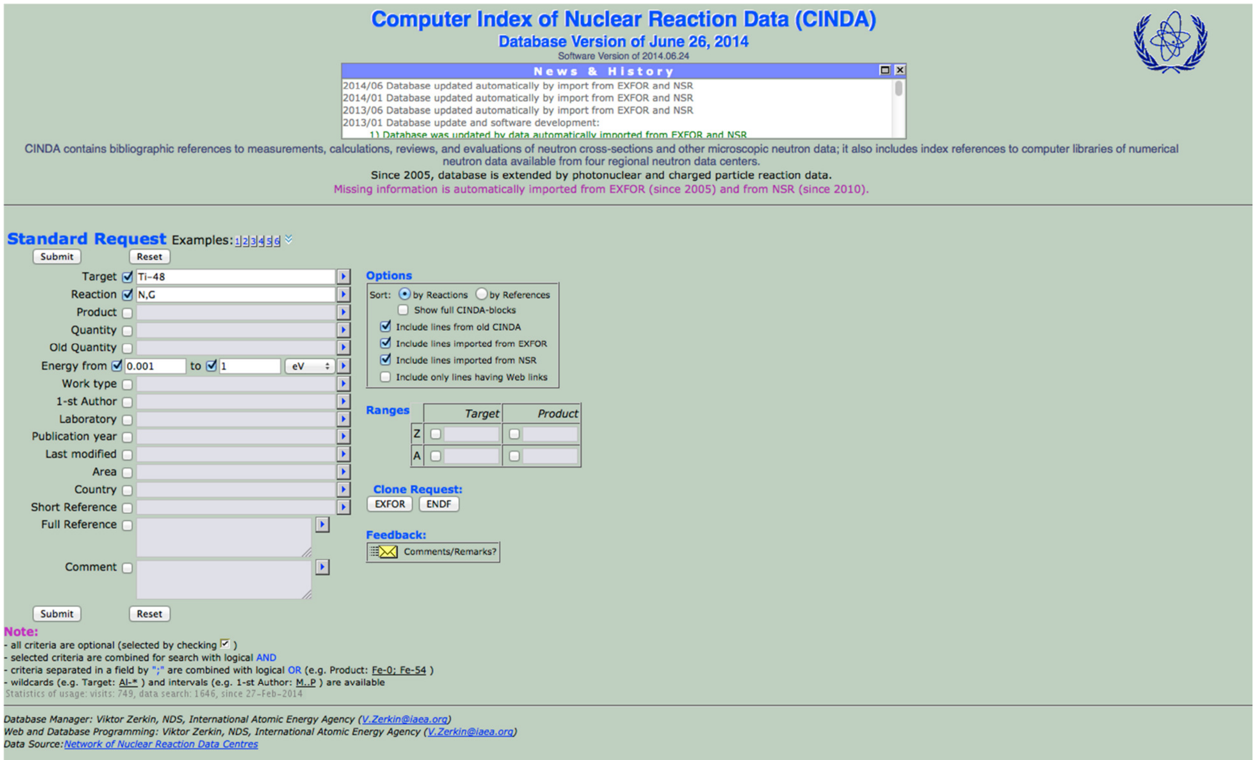
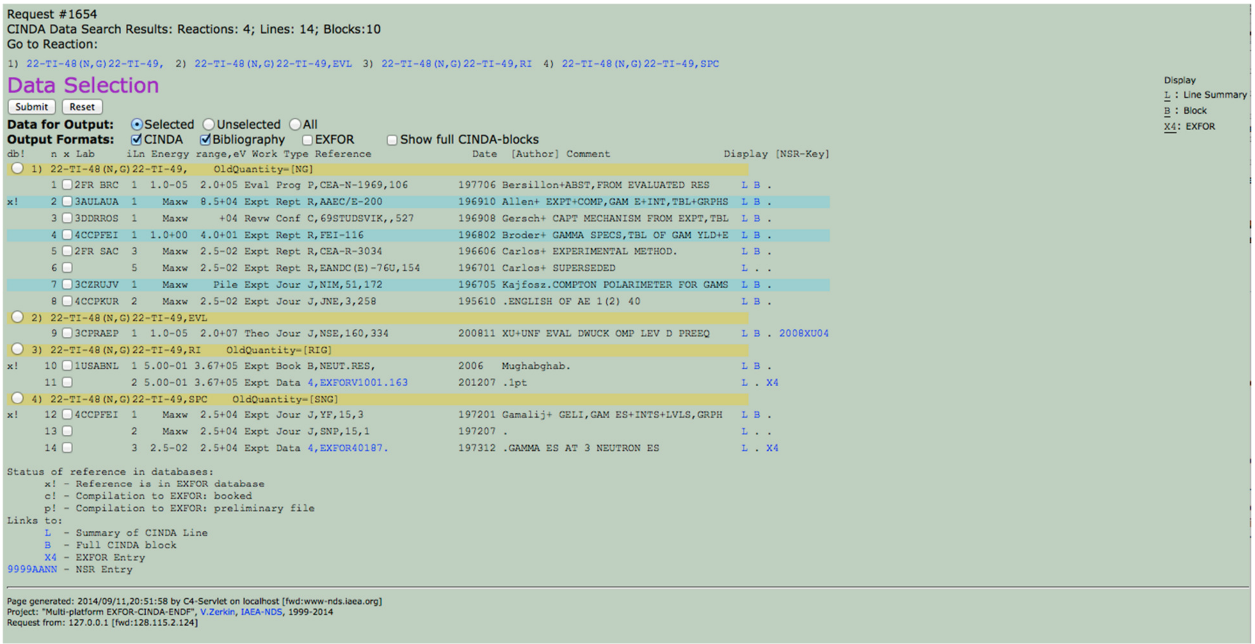


Figure B-4. CINDA Search Results



Request #1654
CINDA Data Search Results: Reactions: 4; Lines: 14; Blocks:10
Go to Reaction:

1) 22-TI-48(N,G)22-TI-49, 2) 22-TI-48(N,G)22-TI-49,EVL 3) 22-TI-48(N,G)22-TI-49,RI 4) 22-TI-48(N,G)22-TI-49,SPC

Display
L : Line Summary
B : Block
X4: EXFOR

Data Selection

Submit Reset

Data for Output: Selected Unselected All
Output Formats: CINDA Bibliography EXFOR Show full CINDA-blocks

```
db!   n x Lab     iLn Energy range,eV Work Type Reference         Date   [Author] Comment                 Display [NSR-Key]
   1) 22-TI-48(N,G)22-TI-49,     OldQuantity=[NG]
      1  2FR BRC   1  1.0-05  2.0+05 Eval Prog P,CEA-N-1969,106    197706 Bersillon+ABST,FROM EVALUATED RES  L B .
x!    2  3AULAUA   1    Maxw  8.5+04 Expt Rept R,AAEC/E-200        196910 Allen+ EXPT+COMP,GAM E+INT,TBL+GRPHS L B .
      3  3DDRROS   1    Maxw    +04 Revw Conf C,69STUDSVIK,,527    196908 Gersch+ CAPT MECHANISM FROM EXPT,TBL L B .
      4  4CCPFEI   1  1.0+00  4.0+01 Expt Rept R,FEI-116           196802 Broder+ GAMMA SPECS,TBL OF GAM YLD+E L B .
      5  2FR SAC   3    Maxw  2.5-02 Expt Rept R,CEA-R-3034        196606 Carlos+ EXPERIMENTAL METHOD.       L B .
      6            5    Maxw  2.5-02 Expt Rept R,EANDC(E)-76U,154  196701 Carlos+ SUPERSEDED                 L . .
      7  3CZRUJV   1    Maxw    Pile Expt Jour J,NIM,51,172        196705 Kajfosz.COMPTON POLARIMETER FOR GAMS L B .
      8  4CCPKUR   2    Maxw  2.5-02 Expt Jour J,JNE,3,258         195610 .ENGLISH OF AE 1(2) 40         L B .
   2) 22-TI-48(N,G)22-TI-49,EVL
      9  3CPRAEP   1  1.0-05  2.0+07 Theo Jour J,NSE,160,334       200811 XU+UNF EVAL DWUCK OMP LEV D PREEQ L B . 2008XU04
   3) 22-TI-48(N,G)22-TI-49,RI     OldQuantity=[RIG]
x!   10  1USABNL   1 5.00-01 3.67+05 Expt Book B,NEUT.RES,         2006   Mughabghab.                L B .
     11            2 5.00-01 3.67+05 Expt Data 4,EXFORV1001.163    201207 .1pt                       L . X4
   4) 22-TI-48(N,G)22-TI-49,SPC     OldQuantity=[SNG]
x!   12  4CCPFEI   1    Maxw  2.5+04 Expt Jour J,YF,15,3           197201 Gamalij+ GELI,GAM ES+INTS+LVLS,GRPH L B .
     13            2    Maxw  2.5+04 Expt Jour J,SNP,15,1          197207 .                          L . .
     14            3  2.5-02  2.5+04 Expt Data 4,EXFOR40187.       197312 .GAMMA ES AT 3 NEUTRON ES  L . X4

Status of reference in databases:
      x! - Reference is in EXFOR database
      c! - Compilation to EXFOR: booked
      p! - Compilation to EXFOR: preliminary file
Links to:
      L  - Summary of CINDA Line
      B  - Full CINDA block
      X4 - EXFOR Entry
9999AANN - NSR Entry
```

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Project: "Multi-platform EXFOR-CINDA-ENDF", V.Zerkin, IAEA-NDS, 1999-2014
Request from: 127.0.0.1 [fwd:128.115.2.124]

Data on the age of cross section data, systems used to measure the differential cross sections, and the number and independence of the evaluations that produced the cross sections are available through these citations.