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Burnup Credit PIRT Report

Brookhaven National Laboratory

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ABSTRACT

When fuel is irradiated in a reactor, the isotopic composition, and therefore the reactivity, of the fuel changes with burnup. Nevertheless, until recently, the NRC's approval of the criticality safety evaluations for commercial spent fuel in casks had been based on analyzing the spent fuel as though it were unirradiated. This "fresh fuel" assumption provided a well-defined, straightforward, and bounding approach that eliminated the need for information on the reactor operating conditions and reactor power history and the core history of individual fuel assemblies. However, while being bounding, the fresh fuel assumption leads to overly conservative cask designs. To address this conservatism, the NRC issued the Spent Fuel Project Office Interim Staff Guidance - 8 Rev.1 based on information available at the time. ISG8R1 allows limited burnup credit (BUC) in the criticality safety analyses of Pressurized Water Reactor (PWR) spent fuel in transport and storage casks. In the same time frame of ISG8R1, the NRC initiated a research program to: clarify guidance on acceptable technical approaches to BUC; develop approaches for expanding the range of BUC; and address regulatory needs for safe, simple, and cost-effective implementation of BUC. As an element in this research program, the NRC commissioned the formation of a Phenomena Identification and Ranking Table (PIRT) panel. Two objectives of the PIRT panel were to 1) identify and rank phenomena, processes, and parameters, that influence the effective neutron multiplication factor (k_{eff}) for irradiated nuclear fuel in transport and storage configurations, and 2) assess the worth of experimental measurement facility classes for validation of calculational methods. Results of the panel's findings are presented in this report.

TABLE OF CONTENTS

Abstract . . . iii

Table of Contents . . . v

List of Figures . . . vii

List of Tables . . . viii

Executive Summary . . . ix

Foreword . . . xi

Acknowledgments . . . xiii

1. INTRODUCTION
- 1.1 Background . . . 1-1
- 1.2 Objective of this PIRT Exercise . . . 1-3
- 1.3 PIRT Overview . . . 1-3
- 1.4 PIRT Panel Membership . . . 1-5
- 1.5 References . . . 1-7

2. PIRT METHODOLOGY
- 2.1 PIRT Assumptions . . . 2-1
- 2.2 Figures of Merit . . . 2-2
  - 2.2.1 Figure of Merit #1 . . . 2-2
  - 2.2.2 Figure of Merit #2 . . . 2-3
- 2.3 Phenomenon Ranking Scale . . . 2-4
- 2.4 Knowledge Assessment Scale . . . 2-5

3. PIRT FOR NEUTRON MULTIPLICATION FACTOR IN A FLOODED CASK
- 3.1 Figure of Merit . . . 3-1
- 3.2 Basis for the PIRT . . . 3-1
- 3.3 Parameters/Phenomena Important to k_{eff} . . . 3-1
  - 3.3.1 Spent Fuel Assembly Materials . . . 3-2
  - 3.3.2 Initial (Fresh) Fuel Enrichment . . . 3-3
  - 3.3.3 Depletion Parameters/Conditions . . . 3-3
  - 3.3.4 Cooling Time . . . 3-4
  - 3.3.5 Burn-up . . . 3-4
  - 3.3.6 Fuel Characteristics . . . 3-6
  - 3.3.7 Cask Characteristics . . . 3-6
  - 3.3.8 Nuclear Data . . . 3-6
  - 3.3.9 Temperature Effects on Criticality in Cask . . . 3-6
- 3.4 Ranking Table for FOM #1 . . . 3-7
- 3.5 Discussion of PIRT Results . . . 3-7
  - 3.5.1 Spent Fuel Assembly Materials . . . 3-7
  - 3.5.2 Initial (Fresh) Fuel Enrichment . . . 3-7
  - 3.5.3 Depletion Parameters/Conditions . . . 3-8
  - 3.5.4 Cooling Time . . . 3-8
  - 3.5.5 Burnup . . . 3-8

3.5.6 Fuel Characteristics . . . 3-8
3.5.7 Cask Characteristics . . . 3-8
3.5.8 Nuclear Data . . . 3-8
3.5.9 Temperature Effects on Criticality in the Cask (Decay Heat) . . . 3-9
3.6 References . . . 3-11

4. PIRT FOR MEASUREMENTS USED FOR CODE AND DATA VALIDATION
4.1 Figure of Merit . . . 4-1
4.2 Basis for the PIRT . . . 4-1
4.2.1 Fresh Fuel Criticals . . . 4-2
4.2.2 Reactivity Worth Experiments . . . 4-2
4.2.3 Subcritical Experiments . . . 4-3
4.2.4 Reactor Criticals . . . 4-3
4.2.5 Radiochemical Assays . . . 4-4
4.3 Ranking Tables for Measurements . . . 4-5
4.3.1 Importance With Respect to Parameters/Phenomena . . . 4-5
4.3.2 Importance for Burnup Credit and Need for Additional Experiments . . . 4-7
4.4 References . . . 4-9

APPENDIX A: LIST OF PRESENTATIONS AT BURNUP CREDIT PIRT PANEL MEETINGS . . . A-1
APPENDIX B: RANKING TABLES FOR FIGURE-OF-MERIT #1 . . . B-1
APPENDIX C: RANKING TABLES FOR FIGURE-OF-MERIT #2 . . . C-1
APPENDIX D: MEMBERS OF THE BURNUP CREDIT PIRT PANEL . . . D-1

LIST OF FIGURES

Figure **Page**

Figure 1-1 Illustration of BUC PIRT Process 1-6
Figure 2-1 Generic Cask Design Example 2-2
Figure 2-2 Cross Section of a 32-Assembly PWR Rail-Type Cask (Example) 2-3

LIST OF TABLES

Table		Page
Table 3-1	Most Important Nuclides in Criticality Calculations	3-2
Table 3-2	Ranking Table for Figure-of-Merit #1	3-10
Table 4-1	Value of Measurement Type for Validating Calculational Methods With Respect to Parameters/Phenomena	4-6
Table 4-2	Evaluation of Experiments/Measurements for Code and Data Validation	4-8

EXECUTIVE SUMMARY

When fuel is irradiated in a reactor, the isotopic composition, and therefore the reactivity, of the fuel changes with burnup. Nevertheless, until recently, the NRC's approval of the criticality safety evaluations for commercial spent fuel in casks had been based on analyzing the spent fuel as though it were unirradiated. This "fresh fuel" assumption provided a well-defined, straightforward, and bounding approach that eliminated the need for information on the reactor operating conditions and reactor power history. However, while being bounding, the fresh fuel assumption leads to overly conservative cask designs.

To address this conservatism, the NRC issued the Spent Fuel Project Office (SFPO) Interim Staff Guidance - 8 Rev.1 (ISG8R1) in July 1999, based on information available at the time. ISG8R1 allows limited burnup credit in the criticality safety analyses of PWR spent fuel in transport and storage casks. Burnup credit (BUC) refers to allowing the criticality safety of spent fuel systems to be evaluated using analysis that considers the reduced reactivity of irradiated fuel. Limited burnup credit refers to limits placed on the licensing basis due to incomplete knowledge and limitations in the ability to model all the relevant physical phenomena in question.

The NRC initiated a research program in April 1999, in order to: (1) clarify guidance on acceptable technical approaches to BUC, (2) develop approaches for expanding the range of BUC, and (3) address regulatory needs for safe, simple, and cost-effective implementation of BUC. As an element in this research program, the NRC commissioned the formation of a Phenomena Identification and Ranking Table (PIRT) panel. Membership of the PIRT panel was drawn from the U. S. and international scientific community, and many of its sixteen members are actively involved in work related to BUC.

The first objective of the PIRT panel was to identify and rank phenomena, processes, and parameters, that influence the effective neutron multiplication factor (k_{eff}) for irradiated nuclear fuel in transport and storage configurations. A second related objective was to assess the worth of experimental measurement facility classes for validation of calculational methods with respect to various categories of phenomena, processes, and parameters that had been established under the first objective.

For the first objective, the PIRT phenomena, processes, and parameters identified by the panel were grouped into nine broad categories: (1) Spent Fuel Assembly Materials, (2) Initial Fuel Enrichment, (3) Depletion Parameters and Conditions, (4) Cooling Time, (5) Burnup, (6) Fuel Characteristics, (7) Cask Characteristics, (8) Nuclear Data, and (9) Temperature Effects on Criticality in the Cask. The importance of each phenomenon was judged relative to a primary evaluation criterion or figure of merit, namely k_{eff} for irradiated nuclear fuel in transport and storage configurations. An importance rank of "High" was assigned if the phenomena had a dominant influence on the evaluation criterion, a rank of "Medium" if it had a moderate impact, and "Low" if it had a small effect.

In addition to the phenomena importance ranking, the panel also ranked the uncertainty in understanding of a particular phenomenon. The panel did so for each phenomenon by assigning a knowledge level for the phenomena to one of three categories: "known" meaning approximately 75-100% of full knowledge and understanding of the phenomenon; "partially known" meaning approximately 25-75% of full knowledge and understanding of the phenomenon; and "unknown" meaning 0-25% of full knowledge and understanding of the phenomenon.

Within the nine broad categories, three subcategories were judged of high importance while at the same time having relatively low knowledge. These subcategories were determined to be: (1) reactivity worth of the spent fuel fission products, (2) axial burnup distribution, and (3) cross sections.

The PIRT process for the second objective proceeded in a manner similar to the first objective but now applied to assessment of the worth of experimental measurement facility classes for validation of calculational methods with respect to various categories of phenomena, processes, and parameters that were previously identified. The methodology being validated refers to both the input data and the computer code models that are used to determine, first the composition of the spent fuel in the cask, and then the k_{eff} of a flooded cask with the fuel present.

The PIRT associated with the second objective was developed through the use of two tables. In one table, the value, or importance, of different measurements was ranked (high, medium, and low) for each of the significant parameters/phenomena. In the other, the importance of the different measurements was ranked specifically for their application to burnup credit with actinides only and with actinides plus fission products together. In addition, the specific question of whether there was a need for additional measurements was asked and the panel provided an answer in terms of a high, medium, or low ranking for each measurement type.

The results are that fresh fuel critical experiments are of value for validating the calculation of cask structural and absorber materials as well as the cask reflector and, to some extent, validating the cross sections for actinides. Reactivity worth experiments and subcritical experiments are in general not very valuable for validating methods. Reactor critical measurements are somewhat valuable for validating the cross sections of actinides. Radiochemical assays are valuable for validating the calculation of both actinide and fission product concentrations.

A second ranking table was developed (1) to evaluate the value, or importance, of measurement types for code and data validation when applied to burnup credit for actinides only and actinides and fission products together, and (2) to determine if there was a need for more experiments in these two categories.

The results for actinide-only burnup credit showed little value for all categories of measurements except reactor critical experiments and radiochemical assays. The corresponding need for additional experiments was obviously low where there was no value. For reactor criticals, there was no need for additional experiments indicating that many already exist, however, there was a strong need for additional documentation to be made available for these measurements so that they could be used by the computational community for cask applications. For radiochemical assays where the value of the measurements was high for actinide-only burnup credit, it was the panel's consensus that more measurements were needed.

The results for "full" burnup credit showed similar trends to that for actinides only. Reactor criticals and radiochemical assays were felt to be important and there was a need for more documentation to be available for reactor criticals and for more measurements using radiochemical assays. The value of reactor criticals was because of their use in validating actinides not fission products. In addition, for fission products, it was clear that reactivity worth experiments were of value and that there was a need for additional experiments, in particular for individual (rather than integral) fission products.

FOREWORD

In 1999 the United States Nuclear Regulatory Commission (NRC) issued guidance for using reactivity credit due to fuel irradiation (i.e., burnup credit) in the criticality safety analysis of spent pressurized-water-reactor (PWR) fuel in storage and transportation packages. This guidance was issued by the NRC Spent Fuel Project Office (SFPO) as Revision 1 to Interim Staff Guidance 8 (ISG8R1) and published in the *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, NUREG-1617 (March 2000). With this initial guidance as a basis, the NRC Office of Nuclear Regulatory Research (RES) initiated a program to provide the SFPO with technical information that would:

- enable realistic estimates of the subcritical margin for systems with spent nuclear fuel (SNF) and an increased understanding of the phenomena and parameters that impact the margin, and
- support the development of technical bases and recommendations for effective implementation of burnup credit (BUC) and provide realistic SNF acceptance criteria while maintaining an adequate margin of safety.

To further the latter objective, RES convened a panel of experts in burnup credit representing different segments of the nuclear industry in the U.S. and oversea. The overall objective is to solicit the panel member's expert's opinions. They do not act as an advisory group to NRC. The idea was to have individual panel members identifies the important phenomena, parameters, and procedures which impact criticality in a flooded cask used for transportation and storage. The panel members were also to consider the uncertainty in the understanding of each phenomenon. Another objective is for panel members to assess the worth of different types of experimental measurements for validation of calculational methods with respect to the phenomena, parameters, and procedures identified as important. These activities would not only further the understanding of BUC but also help prioritize the research needs.

The NRC research on BUC will reduce unnecessary regulatory burden by advancing the use of BUC, while maintaining safety, specifically criticality safety for storage and transportation packages. The use of BUC results in a decrease number of shipments needed, thereby reducing risk from transportation accidents. Lastly, this effort will contribute to making effective, efficient, and realistic regulatory decisions.

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The authors are indebted to the team of Drs. Cecil Parks, John Wagner, and Mark DeHart of Oak Ridge National Laboratory. Under the leadership of Dr. Parks, these experts in burnup credit provided written information to the PIRT panel, made presentations to the panel, and led much of the discussion leading to the PIRT. They facilitated the role of the facilitators. Thanks are also due to the various NRC staff which provided technical and administrative guidance to the panel. Lastly, we appreciated the efforts of Ms. Susan Signorelli who prepared the final version of the report.

1. INTRODUCTION

The United States Nuclear Regulatory Commission (NRC) has commissioned the formation of a Phenomena Identification and Ranking Table (PIRT) panel to identify and rank the phenomena associated with the transport and storage of irradiated (spent) nuclear fuel. The PIRT process is being used to identify and rank the importance of the processes and phenomena that determine the criticality of spent nuclear fuel in the various cask configurations arising during transport and storage. This activity arises from the larger NRC goal of developing the technical bases to support the development of licensing criteria and the application of burnup credit in the calculation of the neutron multiplication constant (k_{eff}) for spent nuclear fuel in cask configurations. The concept of taking credit for the actual reduction in reactivity due to changes in the isotopic composition of the fuel during reactor operations is commonly referred to as burnup credit (BUC).

This report is organized into four sections and contains four supporting appendices. Section 1 - Introduction, identifies the issues associated with the transport and storage of irradiated nuclear fuel, identifies the objectives of the PIRT effort, describes the PIRT process, and identifies the members of the BUC PIRT panel. Section 2 - PIRT Methodology, provides a more detailed description of the PIRT process, as applied to the irradiated nuclear fuel transport and storage issues. Section 3 - PIRT for Neutron Multiplication Factor in a Flooded Cask, summarizes the technical information used by the PIRT panel and summarizes the results of the PIRT activity. Section 4 – PIRT for Measurements, deals with the development of a PIRT that identifies and prioritizes the value of the different types of experiments and measurements with respect to validation of calculational methods for use in BUC applications. Appendix A is a list of presentations made to the panel during their meetings. Appendices B and C provide additional details supporting the summary ranking information presented in Sections 3 and 4, respectively. Each phenomenon's description and rationales for importance ranking, and uncertainty are presented in these appendices. Brief experience summaries for each panel member are provided in Appendix D.

1.1 Background

The applicable background is summarized in two documents. [1-1] [1-2] Much of the following descriptive material is directly extracted from these two sources.

When fuel is irradiated in a reactor, the isotopic composition and therefore the reactivity of the fuel changes. The variation of fuel reactivity with irradiation is primarily governed by the fuel's changing composition of the actinides, fission products, and burnable poison absorbers. Ignoring the presence of burnable poisons, the remaining composition changes will cause the net reactivity of the fuel to decrease. Until now, the NRC's approval of the criticality safety evaluations for commercial spent fuel in casks, including storage and transport casks, has been based on analyzing the spent fuel as though it were unirradiated and without burnable poisons. This "fresh-fuel" assumption has provided a straightforward and bounding approach for showing that spent fuel packages will remain subcritical under normal and accident conditions. The undue conservatism of the fresh-fuel assumption, however, leads to less than optimal design requirements for neutron absorbers and/or spacing of the spent fuel. Also, the fresh fuel assumption provides no insight regarding spent fuel.

The term BUC refers to allowing the criticality safety of spent fuel systems to be evaluated using analysis that considers the reduced reactivity of the irradiated fuel. Allowing credit for fuel burnup necessitates consideration of the fuel operating history, additional validation of calculation methods, consideration of new conditions or configurations, and could involve additional measurements to ensure proper cask loading beyond those contained in the reactor records. The use of a BUC methodology also provides additional understanding of the uncertainties associated with spent fuel. Actinide-only methods of BUC analyze only the effects of changes in the actinide concentrations on fuel reactivity. Full BUC methods analyze the effects of changes in the fission product concentrations as well as those in the actinide concentrations. In commercial power reactor fuels that have achieved most of their intended burnup, actinide effects generally account for about two thirds of the change in reactivity relative to the fresh fuel assumption, with fission products accounting for the remaining one third [1-1].

In the United States, interest in BUC for spent fuel systems has focused mainly on fuel from pressurized water reactors (PWRs) rather than from boiling water reactors (BWRs). This is largely because the smaller pin-array size and correspondingly lower reactivity of individual BWR assemblies, in relation to PWR assemblies, leads to relatively small economic penalties in cask design and capacity when analyzed under the fresh-fuel assumption.

The requirements for transport and dry storage of spent nuclear fuel are included in the Code of Federal Regulations: Part 71 – Packaging and Transportation of Radioactive Material, and Part 72 – Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste. Neither regulation has any specific requirement that would prevent BUC from being implemented in the safety analysis.

Several regulatory bodies in foreign countries have allowed various uses of BUC in wet storage and handling operations. BUC is also being implemented by the NRC for spent fuel storage pools at a number of reactor sites. The French reprocessing program has developed a set of proprietary validation data to support the limited credit needed for shipping modern PWR fuels in the existing fleet of casks. Safety authorities in several countries (e.g., the United Kingdom, Japan, and France), are now working toward similar uses of BUC in transport packages.

The U.S. Department of Energy (DOE) has issued a topical report that proposes a method for incorporating actinide-only burnup effects in the analysis of casks for transporting and storing spent PWR fuel[1-3]. The topical report has gone through two cycles of revisions in response to NRC's review and comments, yet outstanding technical issues and uncertainties have resulted in the NRC not granting the requested approval. The NRC did issue interim staff guidance using information in the DOE topical report and information from other sources[1-4] and incorporated this guidance into an updated NRC Standard Review Plan in March 2000. The NRC's technical review guidance[1-4] [1-5] contains the following recommendations:

1. Limits for the licensing basis

Credit can be obtained for actinides only in UO_2 PWR fuel. The maximum credited burnup is 40 GWd/t. The analyzed cooling time is five years for all fuels cooled longer than five years. There is no credit for fuels exposed to burnable poisons. A loading offset is used for initial enrichments between four and five weight percent.

2. Validation of codes and methods

Isotopic bias and uncertainty are to be derived from applicable fuel assay benchmarks. The bias and uncertainty for k_{eff} are to be derived from benchmark experiments representing the major features of the cask and fuel. Only those nuclides established in the validation process can be used in computing k_{eff}. Bias and uncertainty must be applied in such a manner as to ensure conservatism in the safety analysis. Consideration must be given to additional bias and uncertainty arising from the lack of experiments that are prototypic of spent fuel in a cask.

3. Licensing-basis modeling assumptions

For fuel isotopic calculations, in-core conditions and parameters that maximize k_{eff} in the cask are to be assumed. k_{eff} is to be calculated using models and assumptions that allow adequate representation of important physics, including axial and horizontal burnup profiles within assemblies, the more reactive composition of fuel regions burned in the presence of absorber rods, and local neutron scattering and absorption effects around the most reactive axial regions of the fuel.

4. Loading curve

BUC is available only for assemblies cooled five years or more. The loading curves are to be based on analysis for five years of cooling.

5. Assigned burnup loading value

Administrative procedures, by which the cask user ensures fuel loading is within specification, must be described. A pre-loading measurement is required to confirm the reactor record value of assembly burnup. The assigned burnup loading value must be taken as the confirmed record value of the assembly burnup less combined uncertainties in records and measurement.

6. Estimate of additional reactivity margin

Reactivity margins from actinides and fission products not included in the licensing safety basis are to be estimated. The analysis methods used for estimating margins are to be verified using available experimental data, e.g., fission product assays and worths, and computational benchmarks comparing results to independent methods and analyses. Estimated margins are to be assessed.

1.2 Objective of this PIRT Exercise

The objective of the PIRT panel is to identify and rank the important phenomena or processes relative to the criticality of a flooded cask used for transport and storage of spent nuclear fuel. The PIRT is developed and documented so that it can be used to help guide future NRC-sponsored analytical, experimental, and modeling efforts conducted as part of its program to develop additional technical bases for licensing spent fuel transportation packages that allow credit for decreased reactivity due to fuel burnup. As will be seen below, the PIRT is structured to take into account the important physical phenomena as well as the important experimental information.

1.3 PIRT Overview

The PIRT process has evolved from its initial development and application[1-6] [1-7] [1-8] to its description as a generalized process[1-9]. A PIRT can be used to support several important decision-making processes. For example, the information can be used to support either the definition of requirements for related experiments and analytical tools or the adequacy and applicability of existing experiments and analytical tools. The PIRT methodology brings into focus those phenomena that dominate, while identifying all plausible effects to demonstrate completeness.

A simplified description of the PIRT process, as applied to the development of the BUC PIRT, is illustrated in Fig. 1-1 and described below.

Step 1 is to define the issue that is driving the need, e.g., licensing, operational, or programmatic. The definition may evolve as a hierarchy starting with federal regulations and descending to a consideration of key physical processes.

Step 2 is to define the specific objectives of the PIRT. The PIRT objectives are usually specified by the sponsoring agency. The PIRT objectives should include a description of the final products to be prepared. This is given in Section 1.2 above.

Step 3 is to define the hardware and the hypothetical accident scenario or other assumed conditions for which the PIRT is to be prepared. Generally, a specific hardware configuration and specific scenario are specified. Experience gained from previous PIRT efforts indicates that any consideration of multiple hardware configurations or scenarios impedes PIRT development. After the baseline PIRT is completed for the specified hardware and scenario, the applicability of the PIRT to related hardware configurations and scenarios can be assessed as illustrated in Fig. 1-1. This step is discussed in Section 2.1.

Step 4 is to define the primary evaluation criterion or criteria. The primary evaluation criteria are the key figures of merit used to judge the relative importance of each phenomenon. It must, therefore, be identified before proceeding with the ranking portion of the PIRT effort. It is extremely important that all PIRT panel members come to a common and clear understanding of the primary evaluation criterion, or figure of merit, and how it will be used in the ranking effort. For the BUC PIRT effort, the primary evaluation criterion is derived from a consideration of the physics associated with the transport and storage of spent fuel, namely the degree to which the identified phenomena influence the neutron multiplication factor for irradiated nuclear fuel in transport and storage configurations. The figures of merit for the present PIRT are explained in Section 2.2.

Step 5 is to compile and review the contents of a database that captures the relevant experimental and analytical knowledge relative to the physical processes and hardware for which the PIRT is being developed. Each panel member reviews and becomes familiar with the information in the database. The databases for each of the two figures of merit considered herein are described in Sections 3.2 and 4.2.

Step 6 is to identify all plausible phenomena. In the context of a PIRT, "phenomena" can refer to a physical condition, process, parameter or any other item that helps characterize the figure of merit. The primary objective of this step is completeness. In addition to preparing the list of phenomena, precise definitions of each phenomenon are developed and made available to the PIRT panel to ensure that panel members have a common understanding of each phenomenon. This step results in the framework for the actual PIRT tables that are generated. This step and Steps 7 and 8 are discussed in Sections 3and 4 for the figures of merit.

Step 7 is to develop the importance ranking and associated rationale for each phenomenon. Importance is ranked relative to the primary evaluation criterion adopted in Step 4. For PIRT panels having 6-8 members, ensuing discussions usually lead to a single importance rank for a given phenomenon. For PIRT panels having more members, such as the present one, it has been determined that voting on importance is more practical and usually leads to split decisions. With a large panel, individual members may be experts in some of the phenomena identified but be less familiar with others. To deal with this reality, panel members are informed that they need vote only if they feel they have sufficient understanding of the importance of the phenomenon. Panel members must take care to focus solely on importance relative to the primary evaluation criterion when voting. The degree of knowledge or understanding of the phenomenon is handled separately in the next step.

Step 8 is to assess the level of knowledge, or uncertainty, regarding each phenomenon. This is a new step in the evolving PIRT process and was not included, for example, in a recent generalized description of the PIRT process[1-9]. By explicitly addressing uncertainty, an observed defect of earlier PIRT efforts has been addressed, namely, the tendency of PIRT panel members to assign high importance to a phenomenon for which it is concluded that there is significantly less than full knowledge and understanding.

Step 9 is to document the PIRT results. The primary objective of this step is to provide sufficient coverage and depth that a knowledgeable reader can understand what was done (process) and the outcomes (results). The essential results to be documented are the phenomena considered and their associated definitions, the importance of each phenomenon and associated rationale for the judgment of importance, the level of knowledge or uncertainty regarding each phenomenon and associated rationale, and the results and rationales for any assessments of extended applicability for the baseline PIRT. Other information may be included as determined by the panel or requested by the sponsor.

As presented in Fig. 1-1, the PIRT process proceeds from start to end without iteration. In reality, however, the option to revisit any step is available and is sometimes used in the PIRT development process.

1.4 PIRT Panel Membership

The panel members were selected after considering background related to nuclear fuel characteristics and behavior, cask designs, and technical expertise, e.g., materials science, reactor kinetics and physics, nuclear data, burnup, etc. Representatives of fuel and cask vendors, academia, a national laboratory, a U.S. government agency, and members of the international community were asked to participate.

The panel members participating in the BUC PIRT are:

· George H. Bidinger	Private consultant
· Richard J. Cacciapouti	Duke Engineering & Services
· Jose M. Conde	Consejo de Seguridad Nuclear
· Patrick Cousinou	Institut de Protection et de Sûreté Nucléaire
· Thomas Doering	Electric Power Research Institute
· Peter Grimm	Paul Scherrer Institut
· Hae-Ryong Hwang	Korea Power Engineering Company, Inc.
· Willington J. Lee	NAC International
· Daniel Marloye	Belgonucleaire
· Raymond L. Murray	North Carolina State University
· Jens-Christian Neuber	Siemens AG
· Michaele B. Raap	Battelle Pacific Northwest National Laboratory
· Joseph Sapyta	Framatome ANP
· Daniel A. Thomas	Framatome ANP
· Stanley Turner	Holtec International
· Robert E. Wilson	U.S. Department of Energy

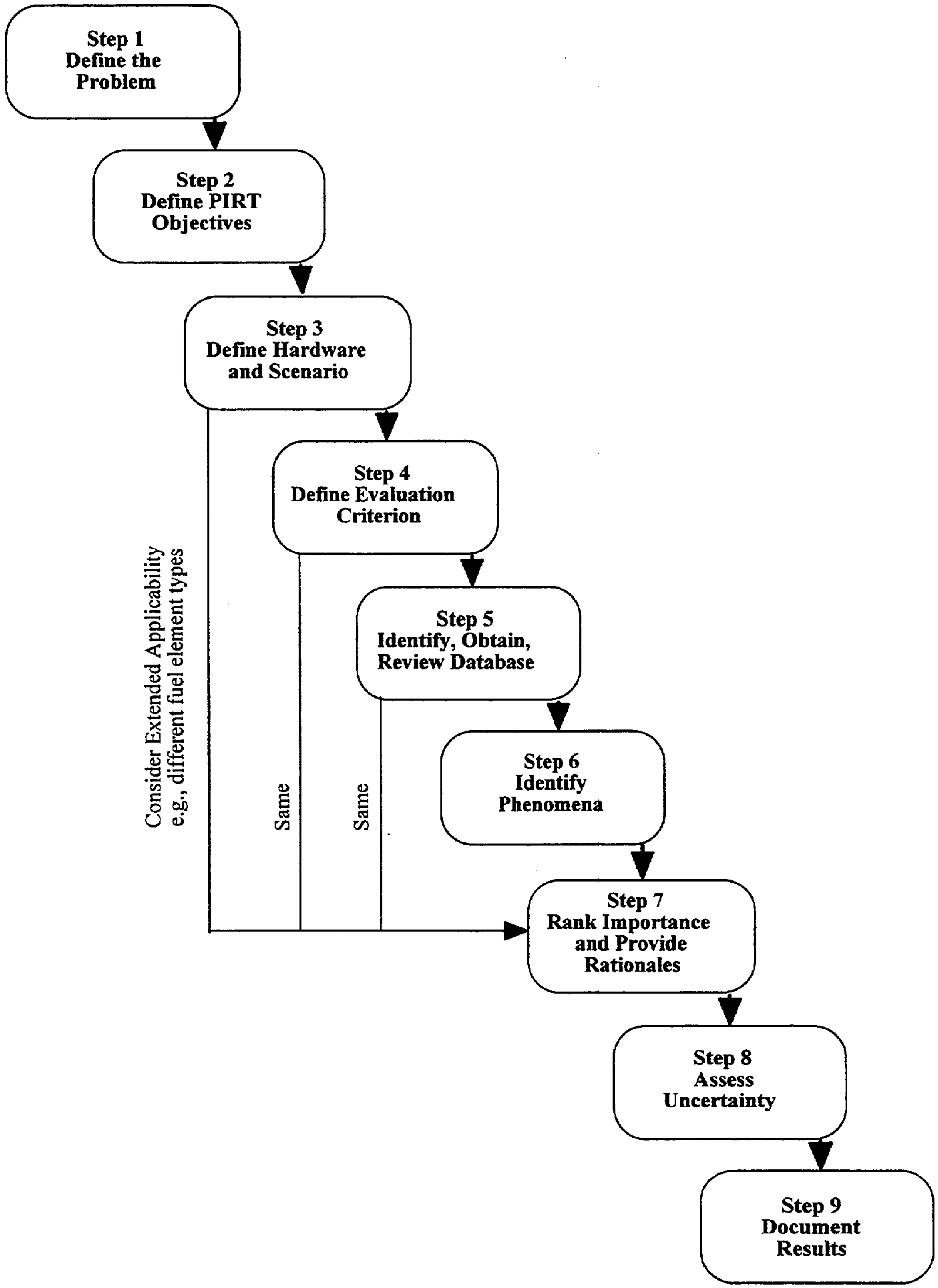

Fig. 1-1 Illustration of BUC PIRT Process

The facilitators for the BUC PIRT panel are David Diamond, Brookhaven National Laboratory and Brent E. Boyack, Los Alamos National Laboratory. Brief experience summaries for each panel member and the panel facilitators are presented at the end of this report in Appendix D.

1.5 References

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1-2 C. V. Parks, M. D. DeHart, and J. C. Wagner, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665, Oak Ridge National Laboratory, February 2000.

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1-5 D.E. Carlson and C.J. Withee, "Regulatory Status of Burnup Credit for Spent-Fuel Storage and Transport Casks," *Trans. Amer. Nucl. Soc.*, November 2000.

1-6 Technical Program Group, "Quantifying Reactor Safety Margins: Application of CSAU to a LBLOCA," NUREG/CR-5249, EG&G Idaho, Inc., 1989.

1-7 Technical Program Group, "Quantifying Reactor Safety Margins: Application of CSAU to a LBLOCA," *Nuclear Engineering and Design, 119,* 1990: B. E. Boyack et al., "Part 1: An overview of the CSAU Evaluation Methodology;" G. E. Wilson et al., "Part 2: Characterization of Important Contributors to Uncertainty;" W. Wulff et al., "Part 3: Assessment and Ranging of Parameters;" G. S. Lellouche et al., "Part 4: Uncertainty Evaluation of LBLOCA Analysis Based on TRAC- PF1/MOD1;" N. Zuber et al., "Part 5: Evaluation of Scale-Up Capabilities of Best Estimate Codes;" I. Catton et al., "Part 6: A Physically Based Method of Estimating PWR LBLOCA PCT."

1-8 R. A. Shaw, T. K. Larson, and R. K. Dimenna, "Development of a Phenomena Identification and Ranking Table (PIRT) for Thermal-Hydraulic Phenomena During a PWR LBLOCA," EG&G Idaho, Inc. report NUREG/CR-5074 (1988).

1-9 G. E. Wilson and B. E. Boyack, "The Role of the PIRT Process in Experiments, Code Development, and Code Applications Associated with Reactor Safety Analysis," *Nuclear Engineering and Design 186*, 23–37 (1998).

2. PIRT METHODOLOGY

As explained in Section 1.3, several important preliminary steps must be completed in advance of the identification and ranking efforts of the PIRT process. The PIRT objective was defined and documented in Section 1.2. Each PIRT is developed based upon a set of assumptions (e.g., hardware configuration and hypothetical accident scenario or assumed conditions), because both the occurrence of phenomena and processes and the importance of phenomena and processes are specific to the hardware and conditions. The hardware configuration and scenario selected for this burnup credit PIRT development are discussed in Section 2.1. As discussed in Section 1.3, the primary evaluation criteria are the figures of merit used by the panel to judge the relative importance of each phenomenon. These are presented in Section 2.2. The phenomenon ranking scale used by the panel is defined in Section 2.3 and the scale used by the panel in developing its assessment of the level of knowledge for each phenomenon is defined in Section 2.4.

2.1 PIRT Assumptions

After being resident in a reactor core for its planned lifetime, a fuel assembly is stored under water in a spent fuel pool at the reactor site. Although fuel assemblies typically remain in the spent fuel pool for a period of years, storage space is limited and the spent fuel must be moved to another location. Agreements are in place with the U.S. government for receipt of the spent nuclear fuel at a permanent repository. Until such time as the permanent repository is ready to accept spent nuclear fuel, some fuel may be moved from the spent fuel pool to a temporary storage site. In either case, transportation packages are used to safely transport and store the fuel. These transportation packages, hereafter referred to as transport casks, must satisfy mandated licensing requirements, thereby assuring public safety. The focus of the burnup credit PIRT panel was transportation of spent nuclear fuel and temporary storage in casks.

Hardware Configuration

The hardware configuration, defined by the NRC, was agreed by the panel to be the basis for the PIRT. The fuel to be transported is intact commercial PWR UO_2 fuel having initial enrichments of 5 w/o or less. All fuel loaded in the transport cask is considered to have been in good condition at the time it was loaded in the cask, i.e., neither damaged nor degraded. The fuel is assumed to have been out-of-reactor and cooling for a period greater than one year but less than 200 years.

Rather than selecting a specific spent fuel transportation cask, a generic cask design as shown in Fig. 2-1 was considered by the panel. The primary functions of the cask are to provide structural integrity to protect the fuel, dissipation of thermal energy, radiation shielding, containment of fission products, and criticality control. The fuel, as stored in the transport cask, is in a dry environment, i.e., initially there is no water in the cask. The inside of the cask is assumed to contain 32 assemblies as shown in Figure 2-2.

Hypothetical Accident Conditions

During transport of the cask from one location to another, an accident is assumed to occur. The details of the accident sequence (e.g., a truck carrying the cask hit by a train) are not specified, nor is the probability of occurrence taken into account. Rather, the NRC specified the immediate post-accident condition of the cask and fuel assemblies contained therein and the panel considered the phenomena arising from those conditions.

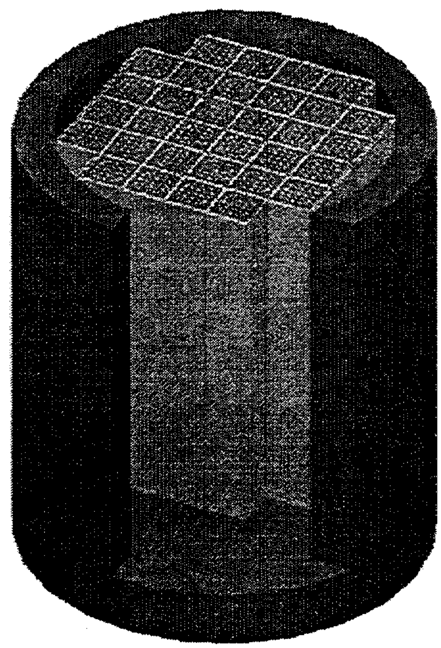

Fig. 2-1 Generic Cask Design Example

The primary change in conditions associated with the accident was that the multiple seals between the fuel and the outside environment were breached. It was further assumed that water from a source such as a river or a lake entered and filled the transport cask (i.e., immersed the fuel assemblies in water). Thus, the fuel assemblies were in a fresh-water-moderated environment as a consequence of the accident.

The panel discussed other potential consequences of the accident, e.g., degraded or damaged fuel configurations. It discussed whether to include fuel damage resulting from the accident in its considerations. It also discussed whether to include changes to the fuel configuration, e.g., fuel rod or assembly bowing, as a result of the accident. In each case, the panel decided that it would not consider these additional potential consequences of the accident.

2.2 Figures of Merit

2.2.1 Figure of Merit #1

The first evaluation criterion or figure of merit (FOM #1) defined by the burnup credit PIRT panel is:

> *The actual neutron multiplication factor (k_{eff}) for irradiated nuclear fuel in transport and storage configurations.*

FOM #1 was chosen because it is the fundamental characteristic of a nuclear fuel configuration for establishing subcriticality. The goal of the PIRT panel with respect to FOM #1 was to (1) identify phenomena, processes and parameters that influence its actual value in the defined fuel cask environment, (2) provide an importance ranking (e.g. high, medium, low) of these phenomena, processes and parameters, and (3) judge the uncertainty associated with each of these phenomena, processes and parameters.

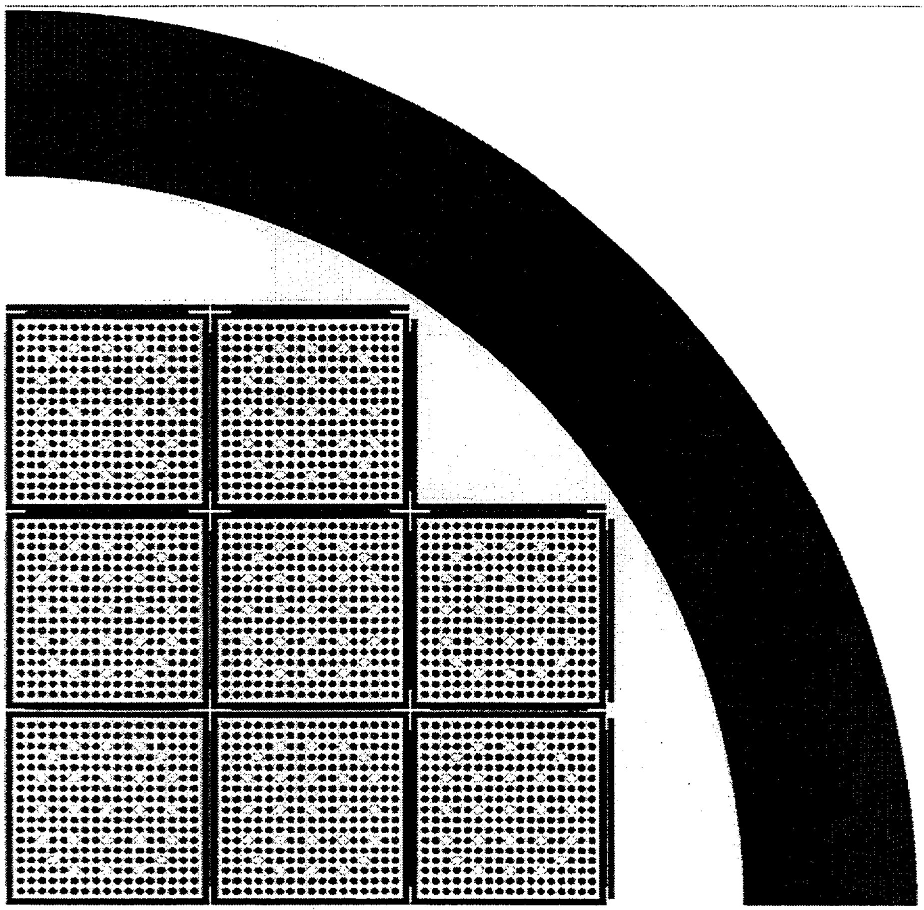

Fig. 2-2 Cross Section of a 32-Assembly PWR Rail-Type Cask (Example)

2.2.2 Figure of Merit #2

The second evaluation criterion or figure of merit (FOM#2) defined by the burnup credit PIRT panel was:

> *The worth of experimental measurement facility classes for validating calculational methods with respect to various categories of phenomena, processes and parameters established in FOM #1.*

FOM #2 was chosen because it will provide guidance in establishing new experimental programs, as well as assessing the value of existing experimental data.

2.3 Phenomenon Ranking Scale

It was decided that the low, medium, and high ranking scheme would be adopted based upon past experience with the PIRT process. In the following ranking discussions, the change in neutron multiplication factor k_{eff} is defined as $\Delta k = (k_{initial} - k_{final})$ where $k_{initial}$ is the initial reactivity and k_{final} is the final reactivity for the corresponding change in the phenomena, process, or parameter of interest over a typical range. Although this discussion relates to Figure of Merit #1, the basic ranking scale can also be extended to Figure of Merit #2.

High = The phenomenon, process or parameter has dominant impact on Figure of Merit #1, the actual neutron multiplication factor (k_{eff}), for irradiated nuclear fuel in transport and storage configurations. The panel discussed and adopted a numerical measure which determined the phenomenon to be of high importance if its reactivity effect Δk is greater than 2.0%. Use of this metric was implicit in the initial ranking efforts, although it was not formally adopted until the panel neared the completion of its ranking efforts. If the importance of the phenomenon is judged to be high, previous PIRT panels have concluded that the phenomenon should be explicitly and accurately modeled in code development and assessment efforts and the phenomenon should be explicitly considered in any experimental programs.

Medium = The phenomenon, process or parameter has moderate influence on Figure of Merit #1 or, numerically, its reactivity effect is $2.0\% > \Delta k > 0.5\%$. If the importance is judged by the panel to be of medium importance, previous PIRT panels have concluded that the phenomenon should be well modeled, but accuracy may be somewhat compromised in code development and assessment efforts. The phenomenon should also be considered in any experimental programs.

Low = The phenomenon, process or parameter has small effect on Figure of Merit #1 or, numerically, its reactivity effect is $\Delta k < 0.5\%$. If the importance is judged to be low by the panel, previous PIRT panels have concluded that the phenomenon should be represented in the code, but almost any model will be sufficient. The phenomenon should be considered in any experimental programs to the extent possible.

A number of earlier PIRT efforts recorded a single importance rank for each phenomenon, with the option of recording any exceptions by a panel member. The assignment of a single importance rank for a given phenomenon was achievable, in part, because the typical panel consisted of only 6-8 members. Such panels were usually able to debate and move to a common view regarding phenomena importance in a timely manner.

The present panel has 16 members (with typically 10 members present at any given session) and the process of debating to a single importance rank for a given phenomenon was usually not possible. Given this situation, it was decided that a vote would be taken and the number of votes for each importance rank reported.

Panel members were asked to vote on only those phenomena for which they have a firm opinion about importance. Generally, the panel member's understanding of importance is understood to arise from direct experience. However, the panel members were free to vote based upon experience in related fields that permitted the panel member to see implications across different fields. The result of not having to vote on all phenomena meant that the vote totals for each phenomenon may be different.

2.4 Knowledge Assessment Scale

The panel also ranked the uncertainty in understanding the effect on the FOM of each phenomenon. The panel did so for each phenomenon by assigning a knowledge level for the phenomena to one of three categories: "known" meaning approximately 75-100% of full knowledge and understanding of the phenomenon; "partially known" meaning approximately 25-75% of full knowledge and understanding of the phenomenon; and "unknown" meaning 0-25% of full knowledge and understanding of the phenomenon. The outcome of the knowledge assessment was recorded and is reported as part of the PIRT tabulation as shown in Sections 3 for FOM #1 and Section 4 for FOM #2.

3. PIRT FOR NEUTRON MULTIPLICATION FACTOR IN A FLOODED CASK

3.1 Figure of Merit

Figure of Merit #1 (FOM #1) is "the neutron multiplication factor (k_{eff}) for irradiated nuclear fuel in storage and transport configurations" with the configurations defined as explained in Section 2.1. This refers to the *actual* k_{eff} of the flooded cask containing spent fuel from a pressurized water reactor. This is not the *calculated* k_{eff}, or equivalently, it is not the calculation of whether the reactivity of the system meets a certain acceptance criterion. Hence, the emphasis in developing the PIRT for this FOM is knowing the condition of the system and its physics, as opposed to the engineering analysis of the system.

3.2 Basis for the PIRT

In order to identify issues that might be important for FOM #1, the panel was sent three documents prior to the first meeting and then several presentations were made to the panel by staff from Oak Ridge National Laboratory (ORNL). One document [3-1] was a comprehensive review prepared by ORNL staff specifically to assist in the PIRT process. The discussion in that report reviews current regulatory practice and perceived industry needs, as "background for prioritizing technical needs that will facilitate safe practice in the use of burnup credit. Relevant physics and analysis phenomenon are identified, and an assessment of their importance to burnup credit implementation is given. Finally, phenomena that need to be better understood for effective licensing, together with technical issues that require resolution, are presented and discussed in the form of a prioritization ranking." Much of the discussion in the report was used as the basis for the oral presentations referred to above. The list of specific presentations provided to the panel prior to developing the PIRT for FOM #1 is given in Appendix A.

A second document received by the PIRT panel was the NRC current guidance on burnup credit based on actinide-only composition [3-2]. It is the expectation that the work of this PIRT panel will be useful in updating this guidance in the future. The panel also received the report "Potential Sources of Experimental Validation for Burnup Credit" [3-3]. This report was not as germane to FOM #1 as it was to the second PIRT which came later to assess the value of measurements, discussed in Section 4.

These documents and presentations provided information for identifying the important phenomena relative to FOM #1 and deciding on their importance and uncertainty. Although some material touched on the calculation of k_{eff} it was recognized that this information was still germane to determining the physics and physical conditions that influence the actual, as compared to the calculated, k_{eff} of the flooded cask. Some of the information discussed prior to actually developing the PIRT tables is discussed in Section 3.3 which identifies categories of the important phenomena/parameters.

3.3 Parameters/Phenomena Important to k_{eff}

The first task of the panel was to identify the parameters/phenomena that influence FOM #1. The panel did this by first identifying broad categories that are relevant and then filling in the detailed parameters/phenomena that are important in each category. The broad categories are:

- Spent fuel assembly materials
- Initial (fresh) fuel enrichment
- Depletion parameters/conditions

- Cooling time
- Burnup
- Fuel characteristics
- Cask characteristics
- Nuclear data
- Temperature effects on criticality in the cask

These are discussed in more detail in the following subsections. In addition, definitions for all parameters/phenomena used in the PIRT are given in Table B-2 of Appendix B.

3.3.1 Spent Fuel Assembly Materials

The spent fuel materials that are potentially important are those that contribute to the reactivity of the system. Hence, the panel chose the following materials in the spent nuclear fuel (SNF) as parameters/phenomena of interest to the PIRT: actinides, fission products, oxygen, residual absorber materials, and non-fuel component compositions (e.g., cladding and assembly hardware).

Part of the basis for this listing came from the background presentation and documentation from ORNL [3-1]; in particular the importance of actinides and fission products. The latter is summarized below:

Several studies have been performed to identify the nuclides that have the most significant effect on the calculated value of k_{eff} as a function of burnup and cooling time [3-4] [3-5] [3-6]. The relative ranking is usually based on the fraction of total absorptions for each nuclide. The adequacy of this simple ranking approach has been confirmed with more rigorous approaches that obtained the actual change in k_{eff} for an infinite lattice of rods based on a change in each nuclide [3-4]. The relative worth of the nuclides will vary some with fuel design, initial enrichment, and cooling time, but the important nuclides remain the same. These studies indicate that the majority of neutron absorption is caused by only a few actinide isotopes and, individually, the fission product isotopes contribute much less to neutron absorption. For cooling times of interest to transport and dry cask storage (1 to 100 years), the relative importance of only a few nuclides changes significantly with cooling time.

Based on these and other studies, the nuclides listed in Table 3-1 are considered to be the prime candidates for inclusion in burnup credit analyses related to dry storage and transport. Obviously, ^{151}Sm (90 yr half-life) and ^{151}Eu are a pair, and ^{151}Eu only needs to be considered if the absorption credit for ^{151}Sm must be maintained.

Table 3-1 Most Important Nuclides in Criticality Calculations

^{234}U	^{235}U	^{236}U	^{238}U	^{238}Pu	^{239}Pu
^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{243}Am*	^{237}Np
^{95}Mo*	^{99}Tc	^{101}Ru*	^{103}Rh*	^{109}Ag*	^{133}Cs
^{143}Nd	^{145}Nd	^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{151}Eu*	^{152}Sm	^{153}Eu	^{155}Gd		

*Nuclides for which measured chemical assay data is not currently available in the U.S.

For the most part, the fission product measurements available in the U.S. for PWR fuel are limited to 3-6 measurements, and prediction methods for these nuclides are not considered to be fully validated. This situation is the major reason that only partial or "actinide-only" burnup credit has been considered by the

DOE [3-7] and the NRC [3-2]. The fission product margin is still present, but since sufficient measured data for isotopic validation do not exist, credit for its negative worth has not been recommended for inclusion in safety analyses.

3.3.2 Initial (Fresh) Fuel Enrichment

This is an obvious category for parameters/phenomena as the initial fissile content along with the burnup determines the fissile content at the time the spent nuclear fuel (SNF) is placed in the cask. However, it is insufficient to consider just the average enrichment of the fuel assembly. Based on this, the panel decided that the parameters/phenomena that needed to be considered were not only the radial-average of the initial enrichment but also the radial variations, i.e., the pin-by-pin variation if any, and the axial variation. The axial variation is found in designs that include axial blankets.

3.3.3 Depletion Parameters/Conditions

The factors that affect depletion are numerous and the panel chose the following for inclusion as parameters/phenomena within this category:

- fuel temperature
- moderator temperature/density
- soluble boron
- specific power
- specific power history
- burnable poison rods
- integral burnable poisons
- control rods

The basis for this selection was again in part the discussion by ORNL and the corresponding report [3-1] which is summarized below:

It is anticipated that burnup credit will be applied for a wide variety of fuel types, each irradiated under a variety of reactor operating conditions. The conditions of interest in a PWR are fuel temperature, moderator temperature and density, soluble boron concentration, specific power and operating history, and the presence of fixed and integral burnable poisons. Although control rods are generally not present in a PWR, in those cases where they are present their effect is similar to that of a fixed poison. If a cask design is intended to accept fuel at different burnups, assumptions that encompass the known variations must be employed in the depletion calculations to ensure that the nuclide content of the fuel is conservatively represented. Several studies [3-4] [3-8] [3-9] [3-10] [3-11] have been performed to assess the effect of depletion modeling assumptions on SNF nuclide predictions. In these parametric analyses, calculated nuclide concentrations were used to calculate k_{eff} for infinite SNF pin lattices and generic casks loaded with SNF. Trends in the neutron multiplication were then examined as a function of each parameter to determine the conservative direction (e.g., high temperature vs low temperature) for that parameter, and the magnitude of the effect over a realistic operating range.

For each parameter studied in Refs. 3-4 and 3-8 to 3-11, the sensitivity of the neutron multiplication to changes in the parameter increases with higher burnups. Furthermore, with the exception of specific power/operating history effects, all of the trends are related to spectral hardening. Spectral hardening results in an increased production rate of plutonium from increased fast neutron capture in ^{238}U. Enhanced plutonium production and the concurrent diminished fission of ^{235}U due to the spectral hardening have the effect of increasing the reactivity of the fuel at discharge and beyond.

In practice, an operational extreme in one parameter may result in an opposite extreme for a coupled parameter. However, simultaneous use of realistic bounding parameter values in a depletion model provides a simple, though conservative, approach to the modeling process since it is unlikely that any fuel would be depleted under all such conditions simultaneously.

3.3.4 Cooling Time

Cooling time was a parameter/phenomenon by itself. The background discussion given to the panel again can be best summarized by quoting from the ORNL report [3-1]:

The five-year cooling time assumed historically in many burnup credit analyses can be traced back to the early policy of the DOE Office of Civilian Radioactive Waste Management not to accept fuel for disposal unless it had a five-year cooling time (sufficient to reduce radiation sources and decay heat values to levels that facilitate higher-capacity cask designs). Fuel discharged from a reactor increases in reactivity for several days due to the decay of short-lived neutron poisons such as ^{135}Xe. After this point, reactivity decreases continuously with time out to about 100 years, at which time it begins to increase. The reactivity continues to increase until a second peak at around 30,000 years, after which time it begins decreasing again [3-6]. The reactivity of the second peak is always less than that occurring at five years. This means that an assumed cooling time for five years is conservative for any cooling time beyond five years. According to Refs. [3-4] and [3-6] the magnitude of the conservatism depends on the initial enrichment and burnup of the fuel. Additional conservatism may be added by basing calculated nuclide compositions on a shorter assumed cooling time period (i.e., cooling periods as short as 1 year).

3.3.5 Burnup

Like initial fuel enrichment, this is an obvious category but care must be taken not to use a simple assembly average value. The panel agreed that the parameters/phenomena that were important were the axial and horizontal burnup within an assembly as well as the burnup distribution across the pin. Furthermore the panel noted (see Table B-2 in Appendix B): "The term burnup has two related meanings: 1) the process of consumption of a substance such as fuel or control material by neutron absorption, and 2) a measure of the specific energy release in fission, with units of [GW-days per metric ton of heavy metal]. In the context of PIRT, k_{eff} is reduced by burnup taking account of the consumption of fissile elements, the buildup of fission products, and structural changes in fuel."

The background given to the panel again can be best summarized by quoting from the ORNL report's [3-1] discussions on axial and horizontal burnup profile. The dynamics of reactor operation result in non-uniform axial burnup profiles in fuel with any significant burnup. At beginning of life in a PWR, the fuel near the axial center of a fuel assembly will deplete at a faster rate than at the ends. As the reactor continues to operate, the flux shape will flatten because of the fuel depletion and fission product poisoning that occurs near the center. However, because of the relatively high leakage near the end of the fuel, burnup will drop off rapidly near the ends. Partial length absorbers or non-uniform axial enrichment loadings can further complicate the burnup profile.

Under a fresh fuel assumption, it is reasonable to assume that fuel is uniformly distributed along the length of a rod, or has discrete axial variations in the case of non-uniform initial loadings. However, for a spent fuel assembly with a reported level of burnup, the burnup value is typically an estimate of the axially averaged burnup. Although it is possible to calculate nuclide concentrations for the average burnup and assume that the material is uniformly distributed along the length of the fuel rod, this is contrary to the reality of the true burnup profile that exists in a spent fuel assembly. The fact that there is a difference between the k_{eff} value calculated assuming an axially varying burnup profile and that calculated assuming a uniform (flat) burnup profile has become known as the "end effect." When assuming an axially uniform distribution of SNF nuclides, the most reactive region of a fuel assembly is at the axial midplane, because leakage increases with distance from the center. However, in reality, the most reactive region of spent fuel is toward the ends, where there is an optimum balance between increased reactivity due to lower burnup and increased leakage due to closer proximity to the fuel ends. [3-4] A fairly extensive review of axial burnup distribution issues that are important to burnup credit criticality safety analyses can be found in the literature [3-12].

Early efforts to address the axial end effect added a margin to compensate for using a uniform profile [3-13]. However, this approach was abandoned when further analyses determined the end effect varies with cask design, the nuclides included in the safety analysis, and burnup. Depending on the cask design, burnup, cooling time, and fuel assembly irradiation history, positive end effects can vary up to several percent [3-14].

As indicated by the above discussion, the most reactive region of the fuel is going to be at a location near enough to the ends to take advantage of the lower burnup, but far enough from the ends that leakage is reduced. This point in an assembly is going to be sensitive to local conditions (within a few mean-free paths) and the local reactivity will not be influenced by the shape of the burnup profile outside this limiting region. In this sense, there is any number of axial profiles that would be equally conservative, if the local conditions at the peak reactivity region are held constant.

Radial variations in the neutron flux, which are due to the core loading as well as leakage at the core periphery, result in a non-uniform horizontal burnup distribution over the radial extent of the reactor core. As the reactor operates, the radial flux shape flattens due to fuel depletion and fission product poisoning. Because of the high leakage near the core periphery, burnup drops off rapidly near the periphery. Ultimately, at the end of a cycle, the individual assemblies located near the center of the core will have a relatively uniform horizontal burnup distribution, while the assemblies near the core periphery may have a significant horizontal variation in burnup [3-15].

To enhance fuel utilization, assemblies are typically relocated within the reactor core between cycle operations. These fuel management practices tend to effectively reduce the horizontal burnup gradient in normal discharged fuel. However, a peripheral assembly discharged after a single irradiation cycle may exhibit a significant horizontal burnup gradient [3-15].

A database containing quadrant-wise horizontal burnup gradients for 808 PWR assemblies (Westinghouse 17×17 and Babcock & Wilcox 15×15) has been prepared, [3-15] and the database has been examined for trends with the number of operating cycles, accumulated burnup, and initial enrichment. No trend with initial enrichment was observed. However, the horizontal gradient was shown to be inversely proportional to accumulated burnup and number of cycles, which are obviously closely related. In other words, the horizontal variation in burnup decreases with increasing burnup.

Based on the horizontal burnup database, Ref. [3-7] has somewhat arbitrarily assigned very conservative bounding values for horizontal burnup gradients to be used in actinide-only burnup credit applications. Further, these gradients are to be applied in conjunction with the most reactive loading configuration. While the proposed approach will conservatively address the concern related to horizontal burnup distributions, it appears to be excessively conservative.

3.3.6 Fuel Characteristics

The fuel characteristics category incorporates considerations that do not fall into the category of fuel composition. Hence, the panel chose the following parameters/phenomena as needing consideration in the PIRT: assembly geometry, cladding hydriding, changes in water-to-fuel ratio in pin cell, and rod bowing. The assembly geometry is clearly an important consideration and includes such items as lattice size, pitch, guide and instrument tubes, etc. The cladding hydriding and changes in the water-to-fuel ratio due to clad creep-down or gap flooding affect the spectrum in the fuel and hence the nuclides generated during burnup.

3.3.7 Cask Characteristics

The previous categories refer to the SNF and it is clear that the characteristics of the cask also need to be taken into account because of their effect on k_{eff} under hypothetical accident conditions. This category and that on temperature effects in the cask (cf Section 3.3.9) define the conditions surrounding the spent fuel. The panel identified the following parameters/phenomena as in this category: cask geometry, cask materials and the water that is assumed to be in the cask. The cask geometry refers to the internals that hold the SNF as well as the surrounding structure. The internals include the basket structure that provides support for the fuel as well as neutron poisons. The outer structure provides shielding, thermal dissipation, and structural integrity. The panel felt that the water should be assumed to be pure and initially at maximum density (corresponding to 4°C). However, as is discussed in Section 3.3.9 the actual temperature of the water due to heating from decay heat is also a consideration.

3.3.8 Nuclear Data

The nuclear data are the fundamental physics parameters that determine reactivity. The panel selected cross sections, decay data, and fission product yields as being the key parameters for the PIRT. The cross sections enter into both the calculation of the reactivity of the cask system and the calculation of the changes in isotopic concentrations of the SNF. The latter is also affected by the decay data and fission product yields. Decay data in this context refers to half-lives and branching ratios that affect the radionuclide inventory in the SNF.

According to the ORNL presentations, actinide cross sections are considered to be well-known as are the majority of fission product cross sections in the energy range of importance to burnup credit. Decay data also have relatively small uncertainties.

3.3.9 Temperature Effects on Criticality in Cask

This category takes into account the effect of decay heat within the cask. The panel separated the parameters/phenomena in this category into the effect on moderator density and the effect on fuel. The temperatures of fuel and moderator are both well known feedback effects that should be considered in criticality calculations.

3.4 Ranking Table for FOM #1

After deciding on the important parameters/phenomena, the panel then voted on the importance and knowledge-base for each of them. Discussion took place prior to each vote. Importance was either high, medium, or low and was in reference to the effect of the parameter/phenomenon on FOM #1 (see Section 2.3). Immediately after each vote a rationale for each of the three choices (high, medium, low) was agreed to by the panel members who voted in that way.

Subsequent to voting on importance, a vote on the knowledge-base for the effect of the parameter/phenomenon on FOM #1 took place. The knowledge level was either known, partially known, or unknown as defined in Section 2.4. Again there was discussion before the vote and after the vote a rationale was agreed upon for each of the three choices.

The result of the voting is shown in the ranking table given in Table 3-2 (and Table B-1 in Appendix B). The table shows each parameter/phenomenon listed. There is a corresponding definition for each parameter/phenomenon given in Table B-2 in Appendix B. The rationales for each of the vote categories (except where there is a zero tally) is given in Tables B-3 and B-4 in Appendix B for importance and knowledge, respectively. Note that the number of votes varies reflecting the abstention or absence of panel members.

3.5 Discussion of PIRT Results

The results of the PIRT can be used to identify the most important considerations. These observations are summarized by category below. The importance rankings and rationales, combined with the knowledge rankings and rationales, have been considered in developing the panel's perspective regarding the important issues affecting cask k_{eff} under hypothetical accident conditions. Of particular interest are parameters/phenomena that have relatively high importance and high uncertainty, i.e., the knowledge base is weak.

3.5.1 Spent Fuel Assembly Materials

The spent fuel assembly materials category consists of five subcategories, only one of which, reactivity worth of the spent fuel fission products, is both important and yet not completely well known. The rationale for high importance is that the reactivity decrease in SNF due to fission products (FP), relative to fresh-fuel, is approximately 15-20% in a cask configuration. However, the panel was divided concerning how great the knowledge of the reactivity worth of these fission products are. Less than half of the panel thought there is a great deal of experience in terms of reactor operations and critical experiments to quantify the reactivity worth of the fission products as well as information from laboratory experiments that indicate that the total FP reactivity is well known. On the other hand, the majority of the panel agreed in general with the minority about knowledge but noted that the experimental evidence has a large uncertainty in some of the individually important FP.

3.5.2 Initial (Fresh) Fuel Enrichment

The initial fuel enrichment category consists of three subcategories, none of which had at the same time a high importance and relatively low knowledge.

3.5.3 Depletion Parameters/Conditions

The depletion parameters/conditions category consists of eight subcategories, none of which had at the same time a high importance and relatively low knowledge. However, both integral burnable poisons and control rods were of medium importance and the knowledge base was only partially known.

3.5.4 Cooling Time

The cooling time category has no subcategories, and it did not have high importance while at the same time having a relatively low knowledge.

3.5.5 Burnup

The burnup category consists of three subcategories, only one of which, axial burnup, had high importance while at the same time having a relatively low knowledge. The rationale for high importance is that several different analysts have shown effects much greater than 2% $\Delta k/k$, and thus it is of high importance according to the ranking scale established in Section 2.3. However, the panel was divided concerning how high the knowledge of the axial burnup is. Less than half of the panel thought that given the profile, the reactivity effect of the axial burnup distribution is well known based on reactor calculations and experience (comparisons of calculations with in-core measurements) and the fundamental physics of the system. On the other hand, the majority of the panel indicated that no experimental benchmarks are available with axial burnup distributions in a cask environment. Also, the panel noted that there is a wide variety of distributions and there are questions regarding proper modeling of the distribution. Therefore due to concerns over the differences between the reactor and a cask environment, this parameter is only partially known.

Horizontal burnup is less important than axial burnup but still of medium importance and because the knowledge base is thought to be only partially known, it too is a parameter of interest.

3.5.6 Fuel Characteristics

The fuel characteristics category consists of four subcategories, none of which had at the same time a high importance and relatively low knowledge.

3.5.7 Cask Characteristics

The cask characteristics category consists of three subcategories, all of which were of high importance but it was only cask materials where the knowledge base was only partially known. The latter is because critical experiments are not available over the entire range of all materials and reactivity worths that are found in cask designs. Consequently, the spectral effects are not necessarily completely covered by the existing critical experiments.

3.5.8 Nuclear Data

The nuclear data category consists of three subcategories, only one of which, cross-sections, was of high importance while at the same time having a relatively low knowledge. The rationale for high importance is that the nuclear cross section data dictates the reactivity, and thus is of high importance. However, the panel was divided concerning how high is the knowledge of the cross sections. Less than half of the panel thought that the measured and experimental data is sufficiently broad and sufficiently known to support an understanding of the effect of cross section on k_{eff}, and thus supports the classification of known. On the other hand, the majority of the panel ranked the knowledge partly known because there are limited benchmark experimental data that tie to reactivity worths for the important nuclides. Thermal neutron data are well known, but the quality of data in the epithermal range requires a classification of partially known.

3.5.9 Temperature Effects on Criticality in the Cask (Decay Heat)

The temperature effects on criticality in the cask category consists of two subcategories both of which were considered to be of low importance.

Table 3-2 Ranking Table for Figure-of-Merit #1

		Importance			Knowledge,		
Category	Parameter/Phenomena	H	M	L	K	PK	UK
Spent Fuel Assembly Materials							
	Reactivity Worth of Actinides in SNF	11	0	0	11	0	0
	Reactivity Worth of Fission Products in SNF	11	0	0	5	6	0
	Reactivity Worth of Oxygen	9	0	0	10	0	0
	Reactivity Worth of Residual absorber materials	0	9	1	8	1	0
	Reactivity Worth of Non-Fuel Component Compositions	8	1	0	10	0	0
Initial (Fresh) Fuel Enrichment							
	Radial-Average Initial Fuel Enrichment	9	0	0	9	0	0
	Radial Variations of Initial Enrichment (pin-to-pin variations)	0	7	2	8	1	0
	Axial Variations of Initial Enrichment (axial blankets)	9	0	0	6	0	0
Depletion Parameters/Conditions							
	Fuel Temperature	0	7	4	10	0	0
	Moderator Temperature/Density	0	10	1	10	1	0
	Soluble Boron	0	8	2	9	1	0
	Specific Power	0	0	11	10	0	0
	Specific Power History	0	2	8	4	5	0
	Burnable Poison Rods	0	11	0	9	1	0
	Integral Burnable Poisons	0	9	0	1	10	0
	Control Rods	0	9	0	6	4	0
Cooling Time	Cooling Time	9	2	0	10	1	0
Burnup							
	Axial Burnup	8	0	0	4	5	0
	Horizontal Burnup	0	7	2	2	7	0
	Burnup Distribution Across Pin	0	0	9	9	0	0
Fuel Characteristics							
	Assembly Geometry	9	0	0	10	0	0
	Cladding Hydriding	0	0	10	10	0	0
	Changes in water-to-fuel ratio in pin cell (e.g., clad creep-down and gap flooding)	0	0	10	10	0	0
	Rod bowing	0	1	11	12	0	0
Cask Characteristics							
	Cask Geometry	10	0	0	9	1	0
	Cask Materials	10	0	0	0	10	0
	Water in the Cask	10	0	0	10	0	0
Nuclear Data							
	Cross Sections	10	0	0	3	7	0
	Decay Data	9	1	0	9	0	0
	Fission Product Yields	10	0	0	10	0	0
Temperature effects on criticality in cask (decay heat)							
	Effect on Moderator Density	0	0	12	11	1	0
	Effect on fuel	0	0	12	12	0	0

3.6 References

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4. PIRT FOR MEASUREMENTS USED FOR CODE AND DATA VALIDATION

4.1 Figure of Merit

The objective of the second portion of the PIRT panel meetings was to assess the value of measurements that can be used to validate the methodology for calculating k_{eff} of a cask used for transport and storage of irradiated nuclear fuel. The figure of merit (FOM) is defined as the worth of experimental measurement facility classes for validating calculational methods with respect to various categories of phenomena, processes and parameters as established in FOM #1. The measurement types of interest are explained in Section 4.2.

The methodology being validated refers to both the data and the computer code models that are used to determine, first the composition of the spent fuel in the cask, and then the k_{eff} of the flooded cask with the fuel present. As a part of the PIRT for FOM #1 the parameters/phenomena important to determining k_{eff} were identified. These parameters/phenomena are also important to the methodology for calculating k_{eff}. The specific parameters/phenomena that need to be validated are discussed in Section 4.3.

The PIRT associated with FOM#2 was developed through the use of two tables. In one, the value, or importance, of different measurements was ranked (high, medium, and low) for each of the significant parameters/phenomena. In the other, the importance of the different measurements was ranked specifically for their application to burnup credit with actinides only and with actinides plus fission products together. In addition, the specific question of whether there was a need for additional measurements was asked and the panel provided an answer in terms of a high, medium, or low ranking for each measurement type.

4.2 Basis for the PIRT

The ideal experimental method for validating the methodology used to calculate k_{eff} for a flooded cask would be to place spent fuel in a cask or cask-like configuration and perform critical experiments. According to the report prepared by the Oak Ridge National Laboratory (ORNL) technical staff supporting the PIRT [4-1], such experiments are extremely challenging because it is very difficult to make even low burnup spent fuel go critical in a controlled manner without first adding some fresh fuel. This is particularly true under cask conditions where external absorbers (e.g., basket material) are present. Spent fuel critical experiments are also complicated by the fact that the fuel samples are highly radioactive, and not as easily manipulated as unirradiated fuel. The expense and complexity of a spent fuel critical is further exacerbated by the need to determine the spent fuel composition by chemical assay (very expensive due to the potentially large number of measurements required) or perform predictive analysis validated against other chemical assay information. No critical experiment using commercial spent fuel in a cask configuration is known to have been performed, although they have been studied [4-2]. Instead, the measurements that can be used to validate the input data and codes used to determine k_{eff} for a cask containing spent fuel are:

- fresh fuel criticals
- reactivity worth experiments - oscillatory methods
- reactivity worth experiments - direct methods
- subcritical experiments
- reactor criticals
- radiochemical assays

These experimental measurements were discussed in presentations made to the panel by the ORNL technical staff supporting the PIRT as well as by panel members. The complete list of presentations is given in Appendix A. In addition, information on the subject was a part of the documentation sent to the panel prior to the first meeting (see Section 3.2). In the following, a brief summary is given of each of these types of measurements along with references for further information.

4.2.1 Fresh Fuel Criticals

According to the ORNL report prepared for the panel [4-1], critical experiments with unirradiated fissile material exist for a wide variety of conditions representative of pin lattices within cask environments[4-3]. The value of these experiments is that they provide validation of particle transport models and cross section data within cask-like conditions. Since these experiments do not contain most of the nuclides that are present in spent fuel and, in the U.S., do not contain the ratios of fissile nuclides actually found in spent fuel, they validate only a part of the calculational methodology. However, proprietary fresh fuel experiments with uranium and plutonium compositions similar to that of typical spent fuel have been performed in France [4-4]. It was the opinion of ORNL and the panel that acquisition of such information would be very valuable for use in actinide-only burnup credit and would assist in validation of casks seeking full burnup credit. Also, proprietary fresh fuel experiments with lattices surrounding cans of fission product solutions have been performed in France and would be excellent experiments to validate methods.

4.2.2 Reactivity Worth Experiments

As stated in the ORNL report prepared for the panel[4-1], to bypass the difficulties associated with using spent nuclear fuel (SNF) assemblies in critical experiments, spent fuel samples (pellets) and samples doped with fission products have been inserted within a fresh fuel system to obtain reactivity worths [4-5]. There are two types of reactivity worth measurements that are typically used. In one the reactivity worth is large enough so that a direct measurement can be taken. In the other the worth is low and oscillatory techniques (i.e. reactivity/period measurements) are used to obtain sufficient accuracy in the measurement. Unless the sample is large enough to provide a significant perturbation to the reference fresh fuel system, the reactivity worth cannot be easily calculated with conventional Monte Carlo codes that are typically used for criticality safety analyses. These experiments can thus provide a means to obtain validation of the reference cross sections used in the criticality analysis, but little else. The French program for burnup credit relies heavily on this approach in conjunction with chemical assay data to demonstrate that the predicted fission product worths are conservative for their codes and that the uncertainty in the fission product cross sections is encompassed by the uncertainty in the prediction of the fission product inventory. Sufficient system perturbation to enable an accurate measure of reactivity worth typically requires isotope concentrations much greater than those present in a spent fuel sample.

When doing reactivity worth measurements with spent fuel samples, the experiment must either have accompanying destructive assays performed, or the fuel design and reactor operation needs to be sufficiently characterized such that an integral benchmark (isotopic prediction and reactivity worth prediction) can be performed to provide a combined validation of both the depletion and criticality methodology.

The ORNL report also provides some information on experiments that have been proposed both in the U.S. and Europe. In Europe the PROTEUS and CERES [4-6] projects are noteworthy. Information on the experiments carried out as part of the PROTEUS project in Switzerland was presented to the panel (see Appendix A). In these experiments reactivity worths will be measured for pressurized water reactor (PWR)

UO_2 and mixed oxide (MOX) fuel with a range of burnups. In addition, reactivity worths of individual nuclides will be made. The lattices in which these worths will be measured will include typical power reactor configurations as well as highly poisoned configurations simulating the harder spectrum that is typical of certain storage configurations. Some of the samples will be subject to chemical assays in order to validate the composition of the spent fuel.

CERES was a joint France and United Kingdom program in which measurements were made on small samples of fuel in zero-power reactors. The fuel came from PWRs and BWRs and included MOX. Additional measurements were done with actinide-only fuel by which is meant fresh fuel with a specified amount of actinides.

4.2.3 Subcritical Experiments

According to the ORNL report prepared for the panel [4-1], subcritical multiplication measurements are an alternative to spent fuel critical experiments that are too difficult to carry out. Calculations can be performed to show the capability to match the predicted multiplication factor to the measured value. As with a spent fuel critical experiment, this validation process would require predictions of spent fuel content prior to the criticality calculations, and would therefore be an integral approach for validation. However, the spectrum should be very similar to that seen in a cask environment and the use of subcritical methods should allow increased flexibility in measuring different configurations. Besides the practical difficulty of handling spent fuel, the performance of subcritical measurements using spent fuel is complicated by the practical difficulties with such measurements in a strong radiation field and the need to interpret k_{eff} from the actual measured quantities [4-7]. The accuracy of subcritical measurements in providing a k_{eff} value for validation is not as good as that provided by a critical experiment. However, the advantage of having an actual spent fuel measurement and its potential to validate SNF cross sections (actual measured quantities are very sensitive to cross-section errors) means that such an experiment could be explored if additional measured data are deemed necessary.

4.2.4 Reactor Criticals

A broad database of critical measurements with partially burned and spent fuel exists in the form of critical configurations for nuclear power plants. Information on this subject was presented by both the ORNL technical support staff as well as panel members. The following is taken from those presentations as well as the report prepared for the panel [4-1].

At a commercial power reactor the conditions of a reactor at a steady state critical condition are known. This includes statepoints in the ascent to power as well as at-power operation. The known reactor statepoint is defined by the core inlet flow and temperature, pressure, boron concentration, control rod positions, and power level. At the beginning of a fuel cycle the core contains a mixture of fresh and burned fuel, and often burnable poisons are present; late in a fuel cycle there is burned and spent fuel, and burnable poisons have typically been depleted.

Like the spent fuel experiments described above, the calculational model of a reactor critical state will require the prediction of spent fuel inventory for each assembly. Given the size of a commercial reactor combined with the variation in operating conditions during a fuel cycle, the task of estimating spent fuel contents at the time of a startup critical or during normal operation is rather formidable. However, all reactors are modeled with codes that follow the behavior of the core throughout the fuel cycle. The methods to accurately predict burn-up and criticality under reactor conditions has been demonstrated by reactor operators for many years. Recently members of the criticality safety community involved in cask studies have shown this to be possible [4-8] [4-9]. In general, cask designers can use fuel burnups as calculated by the reactor operators to demonstrate their ability to predict the critical states.

In addition to the calculation of critical statepoints there are also opportunities to test methods against other startup measurements that are carried out. These include control rod bank worths, differential boron worth, isothermal temperature coefficients, and power distributions.

4.2.5 Radiochemical Assays

According to the ORNL report prepared for the panel, radiochemical assay measurements have been made for select spent fuel nuclides, for both PWR and BWR fuels and an extensive set of references exists [4-1]. The majority of these measurements have been used to determine the biases and uncertainties of computational methods [4-10][4-11][4-12]. In addition, a compilation of sources of radiochemical assay data from these and other sources exists [4-13]. A very limited amount of assay data for mixed oxide spent fuel is available from old U.S. test programs and the more recent ARIANE program coordinated by Belgonucleaire. Sources for additional isotopic assays and an assessment of the completeness of available data describing each set of measurements are also available [4-14].

Chemical assay data have historically focused on the major actinides within PWR spent fuel. The actinides of importance in burnup credit have been measured in 20 or more independent chemical assay evaluations of PWR fuel. For most fission product nuclides important in burnup credit (see Table 3-1), very few assay measurements have been made. In general, the available PWR spent fuel assays correspond to older fuel designs and are limited to less than 40 GWd/t and 3.5 w/o. Additional PWR and BWR spent fuel assays are currently being performed to support DOE programs, but have not been completed and documented at this time.

The use of the chemical assays in the validation process involves a comparison of predicted nuclide concentrations to the measured concentrations. The depletion model is based on the known in-core history for the fuel sample that was characterized. Given a significant number of comparisons, it becomes possible to statistically estimate the bias and uncertainty in the ability to predict the concentration of a given nuclide. Approaches for calculating bias and uncertainty such that one has a reasonable confidence that one can conservatively predict the concentration of a nuclide are given in the literature [4-10] [4-15]. Conservatism is defined in terms of a concentration that has the effect of maximizing k_{eff} for a system. In both of these procedures, calculated biases and uncertainties include any biases and uncertainties inherent in the experimental measurements.

Given the limited number of chemical assays available, and the range of enrichments and burnups represented by these data, it has not been possible to clearly establish trends in biases and uncertainties as a function of the governing parameters. Although some chemical assay data exist for a moderate range of burnups, other factors also vary (i.e., assembly design, operating history, poison concentrations, etc.). Insufficient data prevent the application of a multivariate evaluation. Although additional measurements should be pursued (e.g., for high enrichment and high burnup), the lack of facilities to handle and process spent fuel, combined with the cost of the procedure itself, will limit the number of samples available for validation in the near future.

4.3 Ranking Tables for Measurements

4.3.1 Importance With Respect to Parameters/Phenomena

In order to determine the value of the different measurement types, discussed in Section 4.2, for validating calculational methods, the methods were categorized according to the parameters/phenomena being calculated within the methodology. These parameters/phenomena reflect the panel's deliberations as to what was important in determining k_{eff} for a flooded cask, i.e., the results of the work on FOM #1 given in Section 3. The list of parameters/phenomena considered contains:

- actinide cross sections
- fission product cross sections
- actinide concentrations
- fission product concentrations
- effect of burnable poison rods (BPRs) on SNF compositions
- cask structural and absorber material
- cask reflector material, configuration, and leakage
- SNF assembly interaction (inter- and intra-assembly)
- temperature effects on cross sections (in cask)
- integral absorber effects on SNF compositions

The first vote from the panel was to decide if validation was needed for the parameters/phenomena listed. The vote of the panel is given in Columns 2 and 3 of the ranking table, Table 4-1. (Note that each vote in the table has a rationale that is documented in Appendix C.) Every non-zero vote that validation was needed led to consideration of all the measurement types for that parameter/phenomenon. As a result it was necessary to consider the first 8 of the 10 items listed above. The reason that temperature effects on cross sections was not considered to need validation is that cask licensing calculations are done at cold conditions to be conservative. The reasons that integral absorber effects on SNF compositions did not have to be validated was that the effect was small and the categories of nuclide compositions already implicitly handled the effect. The latter was also the reason that the effect of BPRs on SNF compositions got a strong negative (but not unanimous) vote for whether it needed to be validated.

The remainder of the table contains the vote on the value of each measurement type to the specific parameters/phenomena listed. A high, medium, or low value of importance reflected the panel members' judgement without any quantitative metric for the ranking. The result can be used to reach certain conclusions except for the vote on the value of reactor critical measurements on validating the effect of BPR on SNF reactivity. In this case an almost equal number of the panel voted for both high and low importance with none voting for medium. This conflict reflected a specific disagreement as to the value of these measurements.

As can be seen from the table, fresh fuel criticals are of value for validating the calculation of cask structural and absorber materials as well as the cask reflector and, to some extent, validating the cross sections for actinides. Reactivity worth experiments and subcritical experiments are in general not very valuable for validating methods. Reactor critical measurements are somewhat valuable for validating the cross sections of actinides. Radiochemical assays are valuable for validating the calculation of both actinide and fission product concentrations.

Table 4-1 Value of Measurement Type for Validating Calculational Methods With Respect to Parameters/Phenomena

	Valid-ation Needed?		Fresh Fuel Criticals			Reactivity Worth (Oscillation)			Reactivity Worth (Direct)			Subcritical Measurements			Reactor Critical Measurements			Radiochemical Assays		
Parameter/Phenomenon	Y	N	H	M	L	H	M	L	H	M	L	H	M	L	H	M	L	H	M	L
SNF Nuclide Cross-sections																				
- Actinides	10	0	4	5	0	2	5	2	2	8	0	1	5	4	5	4	1	0	1	9
- Fission Products	10	0	0	0	10	5	3	2	3	6	1	1	6	3	3	4	3	0	0	10
SNF Compositions--Concentration																				
- Actinides	9	0	0	0	9	0	3	6	0	3	6	0	2	7	2	4	3	6	3	0
- Fission Products	9	0	0	0	9	0	3	6	0	2	7	0	1	8	1	4	4	6	2	1
Effect of BPR on SNF Reactivity	3	6	0	0	9	0	0	9	0	0	9	0	0	9	4	0	5	2	1	6
Cask Structural & Absorber Materials	7	2	9	0	0	0	3	6	0	8	1	0	6	3	2	1	6	0	0	9
Cask Reflector	8	1	8	1	0	0	0	9	0	4	5	1	5	3	0	2	7	0	0	9
SNF Assembly Interaction	7	2	0	4	5	0	0	9	0	0	9	0	1	8	4	3	2	0	0	9
Temperature Effects on Reactivity	0	9																		
Integral Absorber Effects on SNF Reactivity	0	8																		

4.3.2 Importance for Burnup Credit and Need for Additional Experiments

The second ranking table was developed 1) to evaluate the value, or importance, of measurement types for code and data validation when applied to burnup credit for actinides only and actinides and fission products together, and 2) to determine if there was a need for more experiments in these two categories. Table 4-2 has these results for the six measurement types discussed above with some embellishments. The measurement types are given in the first column whereas in the previous tables they were given across a row. For reactivity measurements, two additional specific experiment types are added, namely, measurements of individual and integral fission products. The category for reactor critical measurements has been broken into one category for the measurements and another for documentation. This reflects the concern of the panel that data may exist but not be available to the community that needs it for validating methods. The panel voted on the need for additional documentation for reactor critical measurements, although the column heading says "additional experiments."

This vote took into account the discussion that had taken place relative to defining Table 4-1. If a measurement type were important for a parameter/phenomenon that impacts burnup credit then that would also be reflected in Table 4-1.

The panel voted on the importance in terms of high, medium, and low for each measurement type, both for its importance to burnup credit and for the need for additional experiments. The results shown in Table 4-2 are complemented by a rationale for each vote that is given in Table C-2 in Appendix C.

The results for actinide-only burnup credit show little value for all categories of measurements except reactor criticals and radiochemical assays and to some extent, fresh fuel criticals. The corresponding need for additional experiments was obviously low where there was no value. For reactor criticals, there was no need for additional experiments indicating that many already exist, however, there was a strong need for additional documentation to be made available for these measurements so that they could be used by the safety evaluation community for cask applications. For radiochemical assays where the value of the measurements was high for actinide-only burnup credit it was the panel's consensus that more measurements were needed.

The results for "full" burnup credit showed similar trends to that for actinides only. Reactor criticals and radiochemical assays were felt to be important and there was a need for more documentation to be available for reactor criticals and for more measurements providing radiochemical assays. Note, however, that the value of reactor criticals was because of their use in validating actinides not fission products; as evident from the results in Table 4-1. In addition, for fission products, it was clear that reactivity worth experiments were of value and that there was a need for additional experiments, in particular for individual (rather than integral) fission products.

Table 4-2 Evaluation of Experiments/Measurements for Code and Data Validation

	Actinide Only						"Full" Burnup Credit (Actinides + Fission Products)					
	Importance for Burnup Credit			Need for Additional Experiments			Importance for Burnup Credit			Need for Additional Experiments		
Experiment Type	H	M	L	H	M	L	H	M	L	H	M	L
Fresh Fuel Criticals	4	4	1	3	3	3	4	3	2	4	1	4
Reactivity Worth Measurements												
- Oscillation type (small worth)	2	4	3	0	4	4	1	6	2	1	8	0
- Direct measurement (large worth)	1	7	1	0	5	2	2	6	0	2	6	0
- Individual Fission Products							6	2	1	6	3	1
- Integral Fission Products							5	4	0	3	6	0
Subcritical Experiments	1	4	4	1	2	6	1	2	6	1	2	6
Reactor Criticals												
- Measurements	5	3	1	0	2	7	4	5	0	0	2	7
- Documentation				6	2	0				7	1	0
Radiochemical Assays	4	5	0	6	3	0	6	3	0	7	2	0

4.4 References

4-1 C.V. Parks, M.D. DeHart, and J. Wagner, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665, Oak Ridge National Laboratory, February 2000.

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4-11 M. D. DeHart and O. W. Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, September 1996.

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4-15 Office of Civilian Radioactive Waste Management, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," DOE/RW-0472, Rev. 2, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, September 1998.

Appendix A

List of Presentations at Burnup Credit PIRT Panel Meetings

Meeting May 16-18, 2000

"NRC Burnup Credit Research Program," Farouk Eltawila, NRC

"History and Background of Burnup Credit in the US," Cecil V. Parks, ORNL

"Figures of Merit for PIRT," Cecil V. Parks, ORNL

"Overview of NRC Research Program Efforts," Cecil V. Parks, ORNL

"Review of Cask Designs, Analyses, and Pertinent Regulations," John C. Wagner, ORNL

"FOM #1: Proposed List of Parameters and Phenomena Important to Burnup Credit," John C. Wagner and Mark D. DeHart, ORNL

"Review of FOM #2 and ISG-8 Recommendations," Cecil V. Parks, ORNL

"Analyses & Recommendations for Expanded Credit for Cooling Time," John C. Wagner, ORNL

"Analyses & Recommendations for Inclusion of Assemblies Exposed to Burnable Poison Rods," John C. Wagner, ORNL

"Review of the Axial Burnup Distribution Issue and Resolution Recommendations for Actinide-Only Burnup Credit," Mark D. DeHart, ORNL

"Review of the ISG-8 Loading Offset and Resolution Recommendations for Actinide-Only Burnup Credit," Cecil V. Parks, ORNL

Meeting August 22-24, 2001

"Purpose and Objective of PIRT 2," Cecil V. Parks, ORNL

"Review of PIRT 1 Comments Provided by Panelists," Mark DeHart, ORNL

"FOM #1: Review of List of Phenomena Important to BUC," John C. Wagner, ORNL

"Review of FOM #1 and Ranking Process," Cecil Parks, ORNL

"Review of Near-Term Issues from PIRT 1," John C. Wagner, ORNL

"Modeling Issues Important to Burnup Credit," Mark D. DeHart, ORNL

"Relevance of Reactor Information for Licensing of Burnup Credit," Joe Sapyta, Framatome Cogema Fuels

"Spent Nuclear fuel Burnup Verification Devices," Thomas W. Doering, EPRI

"Cask Loading Operations," Willington J. Lee, NAC International

"Axial Burnup Profile Modeling and Evaluation," J-C. Neuber, Siemens Nuclear Power GmbH

"An Empirical Approach for Bounding the Axial Reactivity Effects of PWR Spent Nuclear Fuel," Dan Thomas, et al., Framatome Cogema Fuels

Meeting December 12-14, 2000

"Objectives of PIRT III," Cecil V. Parks, ORNL

"Validation - Approaches, Needs, and Relevance of Experiments," C.V. Parks and M.D. DeHart, ORNL

"Integral and Differential Experiments Pertinent to Burnup Credit," M. Westfall, ORNL

"Proposed Techniques to Assist in Ranking Experimental Data," B.L. Broadhead, ORNL

"Proposed Techniques to Assist in Ranking Experimental Data," I.C. Gauld ORNL

"Applicability of CRC Benchmark Experiments for Burnup Credit Validation," David Henderson Framatome Technologies

"Ranking of PWR Radiochemical Assays," Joe Sapyta and David Henderson, Framatome

"LWR Proteus Phase II: Reactivity Effects and Isotopic Composition of High-Burnup Fuel," Peter Grimm, Paul Scherrer Institut

"Subcritical Multiplication Approach to Direct Measurment of the Neutron Multiplication Constant in Flooded Spent Fuel Casks," David Ebert, NRC

Appendix B

Ranking Tables for Figure-of-Merit #1

This appendix supplements the Phenomena Identification and Ranking Table (PIRT) described in Section 3 and given in Table 3-2. It consists of the definition of the parameters/phenomena in the table and the rationale for each of the votes in those tables. Table B-1 is identical to Table 3-2. Table B-2 gives definitions of terms. Table B-3 provides the rationale for the votes on importance, and Table B-4 the rationale for the votes on knowledge.

Table B-1. Ranking Table for FOM #1: Parameters/Phenomena Important to Neutron Multiplications in a Flooded Cask

Category	Parameter/Phenomenon	Importance			Knowledge		
		H	M	L	K	PK	UK
Spent Fuel Assembly Materials							
	Reactivity worth of actinides in SNF	11	0	0	11	0	0
	Reactivity worth of fission products in SNF	11	0	0	5	6	0
	Reactivity worth of oxygen	9	0	0	10	0	0
	Reactivity worth of residual absorber materials	0	9	1	8	1	0
	Reactivity worth of non-fuel component compositions	8	1	0	10	0	0
Initial (Fresh) Fuel Enrichment							
	Radial-average initial fuel enrichment	9	0	0	9	0	0
	Radial variations of initial enrichment (pin-to-pin variations)	0	7	2	8	1	0
	Axial variations of initial enrichment (axial blankets)	9	0	0	6	0	0
Depletion Parameters/Conditions							
	Fuel temperature	0	7	4	10	0	0
	Moderator temperature/density	0	10	1	10	1	0
	Soluble boron	0	8	2	9	1	0
	Specific power	0	0	11	10	0	0

Table B-1. Ranking Table for FOM #1: Parameters/Phenomena Important to Neutron Multiplications in a Flooded Cask

Category	Parameter/Phenomenon	Importance			Knowledge		
		H	M	L	K	PK	UK
	Specific power history	0	2	8	4	5	0
	Burnable poison rods	0	11	0	9	1	0
	Integral burnable poisons	0	9	0	1	10	0
	Control rods	0	9	0	6	4	0
Cooling Time	Cooling time	9	2	0	10	1	0
Burnup							
	Axial burnup	8	0	0	4	5	0
	Horizontal burnup	0	7	2	2	7	0
	Burnup distribution across pin	0	0	9	9	0	0
Fuel Characteristics							
	Assembly geometry	9	0	0	10	0	0
	Cladding hydriding	0	0	10	10	0	0
	Changes in water-to-fuel ratio in pin cell (e.g., clad creep-down and gap flooding)	0	0	10	10	0	0
	Rod bowing	0	1	11	12	0	0
Cask Characteristics							
	Cask geometry	10	0	0	9	1	0
	Cask materials	10	0	0	0	10	0

Table B-1. Ranking Table for FOM #1: Parameters/Phenomena Important to Neutron Multiplications in a Flooded Cask

Category	Parameter/Phenomenon	Importance			Knowledge		
		H	M	L	K	PK	UK
	Water in the cask	10	0	0	10	0	0
Nuclear Data							
	Cross sections	10	0	0	3	7	0
	Decay data	9	1	0	9	0	0
	Fission product yields	10	0	0	10	0	0
Temperature Effects on Criticality in Cask (Decay Heat)							
	Effect on moderator density	0	0	12	11	1	0
	Effect on fuel	0	0	12	12	0	0

Table B-2. Definition/Description of Parameters/Phenomena

Category	Parameter/Phenomena	Definition
Spent Fuel Assembly Materials		
	Reactivity worth of actinides in SNF	Reactivity worth of actinides in SNF.
	Reactivity worth of fission products in SNF	Reactivity worth of fission products in SNF. This includes fission products that are soluble or volatile.
	Reactivity worth of oxygen	Reactivity worth of oxygen in SNF.
	Reactivity worth of residual absorber materials	Reactivity worth of the residual absorber materials in SNF.
	Reactivity worth of non-fuel component compositions	Effect of non-fuel component compositions (e.g., cladding and assembly hardware) on the reactivity of SNF.
Initial (Fresh) Fuel Enrichment		
	Radial-average initial fuel enrichment	Radial-average (excluding axial blankets) initial fuel enrichment/composition.
	Radial variations of initial enrichment (pin-to-pin variations)	Pin-to-pin variations in initial enrichment.
	Axial variations of initial enrichment (axial blankets)	Axial variations in initial enrichment, including axial blankets.
Depletion Parameters/Conditions		Effect of reactor operating depletion parameters and fuel operating conditions on the reactivity of SNF.
	Fuel temperature	Effect of fuel temperature during depletion on the reactivity of SNF.
	Moderator temperature/density	Effect of moderator temperature/density during depletion on the reactivity of SNF.
	Soluble boron	Effect of soluble boron concentration present during depletion on the reactivity of SNF.
	Specific power	Effect of specific power during depletion on the reactivity of SNF.

Table B-2. Definition/Description of Parameters/Phenomena

Category	Parameter/Phenomena	Definition
	Specific power history	Effect of specific power history (including downtimes, i.e., zero specific-power) during depletion on the reactivity of SNF.
	Burnable poison rods	Effect of burnable poison rod exposure on the reactivity of SNF.
	Integral burnable poisons	Effect of integral burnup poisons on the reactivity of SNF.
	Control rods	Effect of control rod exposure on the reactivity of SNF (both control rod banks and axial power shaping rods).
Cooling Time	Cooling time	Effect of the time period after reactor discharge (cooling time) on the reactivity of SNF.
Burnup		The term *burnup* has two related meanings: (1) the process of consumption of a substance such as fuel or control material by neutron absorption, and (2) a measure of the specific energy release in fission, with units of thermal gigawatt-days per metric ton of heavy metal. In the context of PIRT, k-eff is reduced by burnup, taking account of the consumption of fissile elements, the buildup of fission products, and structural changes in fuel.
	Axial burnup	Axial burnup profile/distribution in the assembly.
	Horizontal burnup	Horizontal or radial burnup gradient in a given assembly.
	Burnup distribution across pin	Radial burnup distribution across fuel pin. Also known as the rim effect, this reflects the buildup of fission products and depletion of fissile material at the outside rim of the fuel pellet.
Fuel Characteristics		
	Assembly geometry	Assembly geometry (e.g., lattice size, pitch, guide/instrument tubes, etc.).
	Cladding hydriding	Presence of absorbed hydrogen in the cladding.
	Changes in water-to-fuel ratio in pin cell (e.g., clad creep-down and gap flooding)	Changes in the water-to-metal ratio in the pin-cell due to changes in the cladding dimensions (either increases or decreases) and/or integrity (gap flooding).
	Rod bowing	Bowing of fuel pins within an assembly caused by reactor core conditions (i.e., not due to postulated accident conditions).
Cask Characteristics		

Table B-2. Definition/Description of Parameters/Phenomena

Category	Parameter/Phenomena	Definition
	Cask geometry	Right circular cylinder containing spent nuclear fuel. The internals of the cask consist of a geometric basket structure that holds the spent nuclear fuel. There are two basic basket structures, flux-trap and lattice basket (no flux trap).
	Cask materials	Structural and basket materials that make up a cask. The internals of the cask consist of a basket structure with various materials that holds the spent nuclear fuel. The basket consists of thermal, structural, and neutron control material.
	Water in the cask	Assumed condition, fresh water in the previous voids in the storage, transport, or disposal package. Fresh water is defined as, full-density water at 4 °C with no impurities.
Nuclear Data		
	Cross sections	Probability of an interaction with a neutron in matter.
	Decay data	The decay constants (half-lives) and branching ratios that affect the radionuclide inventory in irradiated fuel.
	Fission product yields	
Temperature effects on criticality in cask (decay heat)		Effect of temperature increases due to decay heat on the reactivity of SNF.
	Effect on moderator density	Effect of decreased moderator density on the reactivity of SNF.
	Effect on fuel	Effect of fuel temperature on reactivity of SNF.

Table B-3. Ranking Rationale for Importance of Parameters/Phenomena

Category	Parameter/Phenomenon	Importance Ranking Rationale
Spent Fuel Assembly Materials		
	Reactivity worth of actinides in SNF	H: The real essence is that if you were to regard actinides as unimportant, you would have no basis for burnup credit. M: No votes. L: No votes.
	Reactivity worth of fission products in SNF	H: The reactivity decrease in SNF due to fission products, relative to fresh-fuel, is approximately 15-20% in a cask configuration. M: No votes. L: No votes.
	Reactivity worth of oxygen	H: The presence of the oxygen in the fuel has an effect greater than 0.02 delta-k. M: No votes L: No votes
	Reactivity worth of residual absorber materials	H: No votes M: It was originally placed in the fuel and as the fuel burns to a target discharge burnup, the residual remains and is less than 0.01 delta-k for current PWR fuel assemblies. L: Current absorber loadings indicate that the effect is less than 0.005 delta-k.

Table B-3. Ranking Rationale for Importance of Parameters/Phenomena

Category	Parameter/Phenomenon	Importance Ranking Rationale
	Reactivity worth of non-fuel component compositions	H: Due to differences in clad materials (e.g., Zr, SS, ZIRLO), the delta-k can be several percent. The difference can be equal to ~1%. It is suggested that grids are worth ~1% and difference between SS and Zr cladding is ~6-7%. Summary: structural materials are worth ~1%; cladding materials are worth ~6-7%. M: Most of cladding used in current fuel designs are Zr based and thus the effect is of medium importance. L: No votes
Initial (Fresh) Fuel Enrichment		
	Radial-average initial fuel enrichment	H: The reactivity effect of initial enrichment (for a fixed burnup) has been shown to be greater than 0.02 delta-k; initial enrichment is a basic parameter of a loading curve, and thus is of high importance. M: No votes L: No votes
	Radial variations of initial enrichment (pin-to-pin variations)	H: No votes M: Comparisons cited between calculations using actual pin splits and assembly average enrichment show that the effect is less than 0.01 delta-k; this value confirmed based on analysis. L: It is believed that the effect is less than 0.005 delta-k; the intent of pin-split is to even out the burnup, and thus, is notably less than 0.01 delta-k for target burnups. It was confirmed that the effect was less than 0.005 delta-k and clarified that the 0.01 delta-k used for the medium votes was likely quite bounding.
	Axial variations of initial enrichment (axial blankets)	H: The low-enrichment blankets near the ends reduce the reactivity in the low burnup end regions, and thus, reduce the end effect. A number as high as 0.05 delta-k was quoted. M: No votes L: No votes

Table B-3. Ranking Rationale for Importance of Parameters/Phenomena

Category	Parameter/Phenomenon	Importance Ranking Rationale
Depletion Parameters/Conditions		
	Fuel temperature	H: No votes M: The sensitivity is low over the operating range, resulting in a 2% bounding delta-k, which makes it of medium importance. L: The sensitivity reported was conservative (bounding), and thus, a lower delta-k effect (~1%) is expected, resulting in this parameter being of low importance.
	Moderator temperature/density	H: No votes M: The sensitivity is low over the operating range (± 30 K), resulting in approximately 2.5% bounding delta-k, which makes it of medium importance. L: Based on organizational specific studies (including variations in burnup) over an expected operating range (585-615°F), an effect of approximately 1% was observed.
	Soluble boron	H: No votes M: The reactivity effect of variations in the average soluble boron concentration are known to be relatively small (less than 3pcm/ppm), thus it is of medium importance because we have to take into account many different plants (variability). L: For a typical variation of soluble boron concentration, the effect is a few tenths of a percent, which is considered to be low.
	Specific power	H: No votes M: No votes L: Small sensitivity, and thus very minor effect. TRW stated confirmation of presented results.

Table B-3. Ranking Rationale for Importance of Parameters/Phenomena

Category	Parameter/Phenomenon	Importance Ranking Rationale
	Specific power history	H: No votes M: Depending on the time frame of importance, this parameter may vary notably (thinking in terms of disposal or long time frames; and concern regarding BWRs - want to maintain importance in transition to BWRs). L: Based on presented results and discussion (for various conditions), the specific power history results in a relatively small effect on reactivity of SNF.
	Burnable poison rods	H: No votes M: 0.5-3.0% delta-k reactivity effect has been presented by ORNL and confirmed by several other organizations. The range is related to the various cycle exposure assumptions (i.e., one-cycle versus three-cycle exposure). L: No votes
	Integral burnable poisons	H: No votes M: Effect is expected to be similar to that of BPRs (or smaller). L: No votes
	Control rods	H: No votes M: 3% delta-k effect was stated (reference DOE topical, rev. 2), and thus is of medium importance. Could be more important in a small truck-type cask. L: No votes
Cooling Time	Cooling time	H: The cooling time is shown to be ~5% delta-k over the time frame of interest. M: 5% delta-k is not so significant, restrictions on cooling time and loading are not practical. L: No votes
Burnup		

Table B-3. Ranking Rationale for Importance of Parameters/Phenomena

Category	Parameter/Phenomenon	Importance Ranking Rationale
	Axial burnup	H: Several different analysts have shown effects much greater than 0.02 delta-k, and thus is of high importance. M: No votes L: No votes
	Horizontal burnup	H: No votes M: The reactivity effect has been shown to be less than 1% for wet storage and the uncertainty of the impact in a small truck-type cask put this parameter in the range of medium importance. L: The available data, calculational, is very low, and thus indicates that this effect is not significant (important).
	Burnup distribution across pin	H: No votes M: No votes L: This effect has been analyzed and shown to be negligible, and thus, it is of low importance.
Fuel Characteristics		
	Assembly geometry	H: Stated agreement that delta-k differences between varying fuel vendors designs are greater than 0.02 delta-k. A couple of sources of information were cited as support for this classification). M: No votes L: No votes

Table B-3. Ranking Rationale for Importance of Parameters/Phenomena

Category	Parameter/Phenomenon	Importance Ranking Rationale
	Cladding hydriding	H: No votes M: No votes L: Based on results presented or discussed by FCF, ORNL and NRC, the reactivity effect is less than 0.004 delta-k, and thus, is of low importance.
	Changes in water-to-fuel ratio in pin cell (e.g., clad creep-down and gap flooding)	H: No votes M: No votes L: Studies have been performed that consistently demonstrate that the effect is in the range of 0.001-0.003 delta-k, and thus is of low importance.
	Rod bowing	H: No votes M: Increasing the pitch clearly increases the reactivity and the statement that this cannot happen is not acceptable to this panel member. L: Discussion among the panel resulted in general agreement that the fuel rods bow together resulting in little or no change in pitch, and this has a very small associated reactivity effect. This parameter does not include changes in pitch similar to what may be observed in accident conditions. Stated conservatism in analysis that was presented suggests that for a realistic condition, this effect is negligible.
Cask Characteristics		

Table B-3. Ranking Rationale for Importance of Parameters/Phenomena

Category	Parameter/Phenomenon	Importance Ranking Rationale
	Cask geometry	H: The variations in cask geometry (i.e., size and configuration) have significant effect on reactivity, and thus are of high importance. M: No votes L: No votes
	Cask materials	H: The variations in cask materials (i.e., thermal neutron poisons, structural materials, etc.) have significant effect on reactivity, and thus are of high importance. Spectral effects from interaction of materials such as steel and water can have a significant effect, supporting a classification of high importance. M: No votes L: No votes
	Water in the cask	H: Increased moderation due to presence of water increases reactivity. This condition necessitates the consideration of burnup credit for spent fuel storage and transport. M: No votes L: No votes
Nuclear Data		
	Cross sections	H: Nuclear cross section data dictate the reactivity, and thus is of high importance. M: No votes L: No votes
	Decay data	H: Considering the large change in reactivity as a function of decay, this is classified as important M: Rationale for medium vote not captured. L: No votes

Table B-3. Ranking Rationale for Importance of Parameters/Phenomena

Category	Parameter/Phenomenon	Importance Ranking Rationale
	Fission product yields	H: Rationale not captured. M: No votes L: No votes
Temperature effects on criticality in cask (decay heat)		
	Effect on moderator density	H: No votes M: No votes L: The temperature change goes from 0°C to 100°C. Delta-T of 100°C is not going to have a significant positive reactivity effect on reactivity. Cask is not pressurized for this postulated accident, so water boils/freezes at 100/0°C. This also includes cask-loading operations.
	Effect on fuel	H: No votes M: No votes L: Results presented to the panel by ORNL and confirmatory results (FCF) described by panel members suggest that within the range of possible fuel temperatures for a flooded cask (less than 300°F), the reactivity effect is less than ~ 0.004 delta-k.

Table B-4. Ranking Rationale for Knowledge of Parameters/Phenomena

Category	Parameter/Phenomenon	Knowledge Ranking Rationale
Spent Fuel Assembly Materials		
	Reactivity worth of actinides in SNF	K: There is a great deal of experience in terms of reactor operations and critical experiments (thermal systems) to quantify the reactivity worth of the actinides. PK: No votes UK: No votes
	Reactivity worth of fission products in SNF	K: Same as previously specified for actinides. Also, information from laboratory experiments indicates that the total fission product reactivity worth is known to better than 75%. PK: Agree with previous statements that the lumped fission products are known to within 75%, however, there is experimental evidence that there is large uncertainty is some of the individual important fission products. UK: No votes
	Reactivity worth of oxygen	K: Experiments and data for fresh fuel gives us confidence in the reactivity worth in SNF. In addition, there is a large body of evidence in reactors to provide confidence. PK: No votes UK: No votes
	Reactivity worth of residual absorber materials	K: In the U.S. there is significant experience with these materials in operating reactors – these operations require well-known characteristics of the materials, including their concentrations and nuclear parameters. In addition, stated uncertainty in the differential data supports this conclusion. PK: The panel member expressed that there are some observed difficulties with core calculations involving Erbium (referred to some problems at a RBMK reactor in Lithuania). UK: No votes

Table B-4. Ranking Rationale for Knowledge of Parameters/Phenomena

Category	Parameter/Phenomenon	Knowledge Ranking Rationale
	Reactivity worth of non-fuel component compositions	K: Experience with fresh fuel (including critical experiments) provides good level of knowledge for the reactivity worth. PK: No votes UK: No votes
Initial (Fresh) Fuel Enrichment		
	Radial-average initial fuel enrichment	K: There are a large number of critical experiments in which enrichment is varied and compared to calculations – also, there is a wealth of reactor experience with variations in enrichment (e.g., prediction of the lifetime of a reactor/cycle requires a great deal of knowledge regarding the effect of initial enrichment). PK: No votes UK: No votes
	Radial variations of initial enrichment (pin-to-pin variations)	K: In reactor operations, most reactors have internal flux measurements, which are correlated to power. If large uncertainties existed, these effects would be apparent through in-core measurements. Critical experiments have been performed to predict power distribution within an assembly and have correlated well to calculations. Comparison calculations have shown differences of less than 0.01 delta-k, and thus this is well known. PK: Partial knowledge based on the presence and distribution of burnable absorbers. UK: No votes
	Axial variations of initial enrichment (axial blankets)	K: This is a simple spatial calculation in which the enrichments are well know, and thus the knowledge of the reactivity effect of this parameter is well known. Knowledge is also high due to reactor core following experience. PK: No votes UK: No votes

Table B-4. Ranking Rationale for Knowledge of Parameters/Phenomena

Category	Parameter/Phenomenon	Knowledge Ranking Rationale
Depletion Parameters/Conditions		
	Fuel temperature	K: Given the temperature, the effect of temperature on the reactivity of SNF can be accurately calculated. PK: No votes UK: No votes
	Moderator temperature/density	K: If we can accurately do core calculations, which include the variations in moderator temperature, we can accurately calculate the reactivity for SNF. PK: Historically, differences between measurements of moderator temperature coefficients and calculations have been observed. Consequently, partial knowledge is considered appropriate. UK: No votes
	Soluble boron	K: Consistent reporting of analyses from independent sources supports the conclusion that this effect is well known. PK: The manner in which we justified the importance on the operating range variations is based on calculational approximations. UK: No votes
	Specific power	K: Confirmation from multiple sources and reactor operation histories provides confidence in this knowledge. PK: No votes UK: No votes
	Specific power history	K: Based on presented results and discussion, the effect of specific power history appears to be known. PK: There does not appear to be any specific trends, suggesting that there are aspects that we do not understand well. Small effects are difficult to quantify, and thus confidence is not high. UK: No votes

Table B-4. Ranking Rationale for Knowledge of Parameters/Phenomena

Category	Parameter/Phenomenon	Knowledge Ranking Rationale
	Burnable poison rods	K: Body of data from reactor operations (from different fuel vendors) with and without BPRs provides confidence in our knowledge of the effect of BPRs. PK: The knowledge base for justifying the "known" vote is based on reactor operations (not SNF casks) and proprietary data. There are new assembly designs that use BPRs that have not been analyzed (e.g., CE fuel assembly). UK: No votes
	Integral burnable poisons	K: Core-follow data shows excellent agreement between predicted data and actual core data (may be related to proprietary data). The poison is depleted during the first half of the first cycle – the resulting impact at discharge is small. PK: Based on the available information (proprietary chemical assays) we have partial knowledge, but require additional information to enhance understanding. Uncertainty exists on the direction of the reactivity change due to integral Gd & Er absorbers. More information may be available for BWR fuel. UK: No votes
	Control rods	K: Based on reactor follow data, reasonable confidence in the rod worths may be achieved. Predictability should be similar to that of BPRs. PK: Analyses were not deemed to be complete to support a full understanding of the phenomena. Additionally, there seems to be a lack of available/evaluated benchmark data related to the use of control rods. UK: No votes

Table B-4. Ranking Rationale for Knowledge of Parameters/Phenomena

Category	Parameter/Phenomenon	Knowledge Ranking Rationale
Cooling Time	Cooling time	K: Based on stated reactivity worth experiments performed in the U.K., which resulted in consistent agreement within 5% with calculated results, one would not expect differences higher than 10% over the time period of interest. Sensitivity studies (DOE) of delta-k with time over a longer time period than considered here demonstrated very small differences in reactivity. PK: Experimental data for the justification is limited and not freely available. UK: No votes
Burnup		
	Axial burnup	K: Given the profile, the reactivity effect of the axial burnup distribution is well known based on reactor calculations and experience (comparisons of calculations with in-core measurements) and the fundamental physics of the system. PK: No experimental benchmarks are available with axial burnup distributions in a cask environment. Also, there is a wide variety of distributions and there are questions regarding proper modeling of the distribution. Due to concerns over differences between the reactor environment and a cask environment, this parameter is partially known. UK: No votes
	Horizontal burnup	K: Given the problem as stated (i.e., known profile), this is a simple physics problem that we have a good deal of knowledge with. PK: Lack of measurements (not included in core follow data) and variations in opinions/discussion, means the knowledge of this effect is partially known. UK: No votes
	Burnup distribution across pin	K: Agreement between experimental and calculational data supports an assessment of well-known. PK: No votes UK: No votes
Fuel Characteristics		

Table B-4. Ranking Rationale for Knowledge of Parameters/Phenomena

Category	Parameter/Phenomenon	Knowledge Ranking Rationale
	Assembly geometry	K: Many years of experience in reactor operations and spent fuel storage pools support a classification of well known. Critical experiments including variations in pitch and rod diameters provide additional support. PK: No votes UK: No votes
	Cladding hydriding	K: Hydrogen is a moderator that is well known. Additionally, the results presented as justification that the effect is of low importance are applicable to demonstration that the effect is known. PK: No votes UK: No votes
	Changes in water-to-fuel ratio in pin cell (e.g., clad creep-down and gap flooding)	K: Fresh fuel critical experiments with variations in the water-to-fuel ratio for pin cells provide support for classifying this as well-known. Additionally, results cited as a function of burnup and the large body of reactor operating data support this classification. PK: No votes UK: No votes
	Rod bowing	K: Critical experiments have been successfully compared to calculated results for a number of different lattices providing support for a classification of known. (Note: this rationale is based on the fact that the rod bowing configuration is known) PK: No votes UK: No votes
Cask Characteristics		
	Cask geometry	K: Fresh fuel critical experiments with variations in geometric configuration support the classification of known. However, materials of which those geometries are composed are not necessarily verified by experiment. PK: Difficulty in separating geometric effects from poison materials leads to the classification of partially known. UK: No votes

Table B-4. Ranking Rationale for Knowledge of Parameters/Phenomena

Category	Parameter/Phenomenon	Knowledge Ranking Rationale
	Cask materials	K: No votes PK: Critical experiments are not available over the entire range of all materials and reactivity worths that are found in cask designs, and thus this is classified as partially known. Consequently, the spectral effects are not necessarily completely covered by the existing critical experiments. UK: No votes
	Water in the cask	K: A large number of critical experiments involving water moderation are available and have been used to demonstrate our knowledge of the effect of water moderation. PK: No votes UK: No votes
Nuclear Data		
	Cross sections	K: The fact that the measured and experimental data is sufficiently broad and sufficiently known to support an understanding of the effect of cross section on keff, and thus supports a classification of known. PK: There are limited benchmark experimental data that tie to reactivity worth for the important nuclides. Thermal data are well known, but the quality of data in the epi-thermal range requires a classification of partially known. UK: No votes
	Decay data	K: Results were cited supporting the classification of well-known. PK: No votes UK: No votes
	Fission product yields	K: Rationale based on post-irradiation-examination data PK: No votes UK: No votes
Temperature effects on criticality in cask (decay heat)		

Table B-4. Ranking Rationale for Knowledge of Parameters/Phenomena

Category	Parameter/Phenomenon	Knowledge Ranking Rationale
	Effect on moderator density	K: We have experimental data to back up our ability to calculate k as a function of water density. We also have computer models that are able to make the same predictions. PK: There are a wide variety of unknown configurations. UK: No votes
	Effect on fuel	K: Reactor operations (start-up) experience and critical experiment at elevated temperatures suggest adequate knowledge to classify this effect as known. PK: No votes UK: No votes

Appendix C

Ranking Tables for Figure-of-Merit #2

This appendix supplements the Phenomena Identification and Ranking Table (PIRT) described in Section 4 and given in Tables 4-1 and 4-2. It consists of the rationale for each of the votes in those tables. These are presented as footnotes to the entries in the tables.

Table C-1. Value of Measurement Type for Validating Calculational Methods with Respect to Parameters/Phenomena

	Validation Needed?		Fresh Fuel Criticals			Reactivity Worth (Oscillation)			Reactivity Worth (Direct)			Subcritical Measurements			Reactor Critical Measurements			Radiochemical Assays		
Parameter/Phenomenon	**Y**	**N**	**H**	**M**	**L**	**H**	**M**	**L**	**H**	**M**	**L**	**H**	**M**	**L**	**H**	**M**	**L**	**H**	**M**	**L**
SNF Nuclide Cross-Sections																				
- Actinides	10[1]		4[2]	5[3]		2[4]	5[5]	2[6]	2[7]	8[8]		1[9]	5[10]	4[11]	5[12]	4[13]	1[14]		1[15]	9[16]
- Fission Products	10[17]				10[18]	5[19]	3[20]	2[21]	3[22]	6[23]	1[24]	1[25]	6[26]	3[27]	3[28]	4[29]	3[30]			10[31]
Temperature Effects on Cross-Sections (in Cask)		9[32]																		
SNF Compositions (Concentrations)																				
- Actinides	9[33]				9[34]		3[35]	6[36]		3[37]	6[38]		2[39]	7[40]	2[41]	4[42]	3[43]	6[44]	3[45]	
- Fission Products	9[46]				9[47]		3[48]	6[49]		2[50]	7[51]		1[52]	8[53]	1[54]	4[55]	4[56]	6[57]	2[58]	1[59]
Effect of BPR on SNF Compositions	3[60]	6[61]			9[62]			9[63]			9[64]			9[65]	4[66]		5[67]	2[68]	1[69]	6[70]
Integral Abs. Effects on SNF Compositions	0	8[71]																		
Cask Structural and Absorber Materials	7[72]	2[73]	9[74]				3[75]	6[76]		8[77]	1[78]		6[79]	3[80]	2[81]	1[82]	6[83]			9[84]
Cask Reflector Material(s) Configuration & Leakage	8[85]	1[86]	8[87]	1[88]				9[89]		4[90]	5[91]	1[92]	5[93]	3[94]		2[95]	7[96]			9[97]
SNF Assembly Interaction (Inter- and Intra-assembly)	7[98]	2[99]		4[100]	5[101]			9[102]			9[103]		1[104]	8[105]	4[106]	3[107]	2[108]			9[109]

Footnotes for Table C-1

1 Yes – Actinides are the key parameter affecting k-eff.
2 H – Fresh fuel criticals are unambiguous and clean. They are also good benchmarks (i.e., show a good correlation) at low burnup in ORNL's sensitivity studies.
3 M – Fresh fuel criticals do not include the full set of actinides in spent fuel – they contain a limited set of actinides. Fresh fuel criticals are of high value for validating the major nuclides that are present in fresh fuel experiments.
4 H – It is a rigorous assessment of the cross section if the composition is well known. Also, measurements can be done in a spectrum tailored to look like a specific application, i.e., a spent fuel cask.
5 M – Considerable value in the case of a lack of direct (fresh fuel) validation where a nuclide is not present.
6 L – Applicability for code validation is limited as a sole source since these experiments cannot be used directly in validation of criticality codes.
7 H – Same as for oscillation (high), but easier to obtain. Unlike oscillation, it can be directly used in validation of the codes/methods used for cask analysis.
8 M – Same as for oscillation (medium); in most cases, the composition of the material cannot be accurately quantified. Can be directly used in validation.
9 H – Noise analysis experiments are very sensitive to cross sections.
10 M – Acceptable method for assessment of cross section data on "obscure" nuclides, especially in light of reduction of capability for critical experiments. Difficult to use in direct validation.
11 L – Difficult to use in direct validation.
12 H – All isotopes are available, can be used in direct validation, and many measurements are available or potentially available.
13 M – Readily available, but the results are somewhat ambiguous (too many variables), i.e., does not directly validate cross sections. Not as clean as fresh fuel criticals. Spectra are different between hot reactor core and cask.
14 L – Only provides integral data for some nuclides.
15 M – Provides some information which can be of value when taken with other experiments.
16 L – Does not give direct assessment of cross section data – provides anecdotal information only. Too many other complications. Difficult to separate out effects of nuclides due to isotopic changes during depletion.
17 Yes – Fission products are necessary for a realistic estimate of k-eff, i.e., they are the second key phenomenon.
18 L – Fission products are not present in fresh fuel (except for Gd and other burnable poison materials).
19 H – They provide data lacking due to the lack of fresh fuel criticals for validation. Also same as for actinides.
20 M – Same as for actinides.
21 L – Same as for actinides. Monte Carlo criticality codes cannot be validated using this type of experiment.
22 H – Same as for actinides.
23 M – Same as for actinides.
24 L – Typically small amounts are used, causing large uncertainty. Large samples will perturb the spectrum.
25 H – Same as actinides.
26 M – Same as actinides.
27 L – Same as actinides. Would be of more value if used to validate the lumped worth of all fission products.
28 H – Same as actinides. Number of experiments available may show that there are no compensating effects.
29 M – Same as actinides, except for reference to fresh fuel. Gives only integral information on cross sections. Does provide information on overlapping resonances.
30 L – Too different from cask configurations. Fission product worth is small relative to actinides. Some fission products do not exist in appreciable

quantities in a reactor.

31 L – Same as for actinides.

32 No – The conservative (cold) approach is dictated for licensing.

33 Yes – The actinide compositions (computed by depletion codes) are a key item in accurately estimating k-eff.

34 L – This class of experiments does not apply to the prediction of spent fuel actinide compositions, and are therefore not applicable.

35 M – Provides a possible means for validating codes with experimental measurements if spent fuel is used in the samples.

36 L – Same as medium, also there is no direct measurement – approach is integral.

37 M – Same as for oscillations (medium). Also, larger effect on reactivity in larger samples is closer to applications modeled.

38 L – Accuracy is not as good as that of oscillation experiments. Also provides no direct measurement.

39 M – Subcritical methods are on the verge of being mature and applicable.

40 L – Subcritical methods are not yet fully developed. Practicalities in an operating commercial plant would be problematic. k-eff value is an indirect measurement and gives no direct information on compositions.

41 H – Good source of information on partially burned fuel. Provides a validation of the actual quantity that the criticality codes will predict (k-eff) for the integral system.

42 M – Not a clean experiment, and has a lot of variables, but still a solid source of reactivity information.

43 L – Cannot isolate actinides from fission products, i.e., not a direct and interpretable measurement of actinide concentrations.

44 H – Only direct measurement of isotopic concentrations.

45 M – Samples come from a complex configuration, and the data is not as clean as it could be (unknown local burnup, history uncertainties, etc.). Validation of criticality codes are based on k. Not directly related to total core inventory.

46 Yes – Same as above, but for FPs.

47 L– Same as above for FPs

48 M – Same as for actinides

49 L – Same as for actinides

50 M – Same as for actinides

51 L – Same as for actinides

52 M – Same as for actinides

53 L – Same as for actinides

55 H – Same as for actinides

55 M – Same as for actinides

56 L – Same as for actinides

57 H – Same as for actinides

58 M – Same as for actinides, but also more uncertainty in separation of fission products.

59 L – Large variability in fission product measurements, indicating problems in measurements relative to actinides.

60 Yes – Has a significant (~1%) effect and should be checked.

61 No – This is implicit in validation for compositions ranked earlier. Also, there is sufficient knowledge to make conservative assumptions for treatment of BPRs.

62 Low – Not applicable

63 L – Not applicable; too small to measure

64 L – Same as for oscillation

65 L – Not applicable

66 H – 1% effect on reactivity (in a fully loaded cask) will be picked up in a core model
67 L – You get no information on the compositions from this type of experiment; there is no explicit information
68 H – Pu production from BPRs should be seen in chemical assays.
69 M – There is potential for useful information from experiment.
70 L – Delta in composition will be small (perhaps even smaller than the measurement uncertainty)
71 No – Too small to see in measurements. Also implicit in validation of compositions ranked earlier.
72 Yes – Used for reactivity suppression. Used in safety analysis and must be affirmed.
73 No – Not specific to BUC – always needed. Also done for spent fuel pools.
74 H – Fresh fuel criticals are clean and adequate for measuring the worth of materials external to assembly.
75 M – Oscillation experiment would be ideal for certain absorbers to understand their effect on the spectrum.
76 L – Monte Carlo validation is difficult with oscillation experiments. Measurements would not be representative of energy range or scattering kinematics of cask.
77 M – Measurements have been done that show clear Δk differences.
78 L – Such a measurement would really be a fresh fuel critical.
79 M – Measurements can be made in a cask, and can be made cleanly and easily with fresh fuel.
80 L – Concur with medium when fresh fuel is used, but a spent fuel experiment with have too small a value of k to get accurate subcritical measurements. Subcritical measurements have potential to adversely impact the subcritical margin due to their larger uncertainty and their basis (measurement of subcritical state rather than critical state).
81 H – Reactor is representative of cask conditions due to fission products and depleted actinides.
82 M – Agree with high, but mismatch of geometry and material concentrations.
83 L – Configuration and material densities in cask are more important than the composition of the fuel.
84 L – Absolutely, positively not applicable.
85 Yes – Safety credit is taken for these parameters, and must be validated. Materials and geometry strongly influence k-eff. Shielding is an important aspect of BUC casks.
86 No – This is not solely a BUC issue. It is generic to criticality safety in any cask.
87 H – Materials can be easily mocked up in a fresh fuel critical, and the spectrum can be tailored to simulate spent fuel
88 M – Not sure the spectrum can be tailored to spent fuel spectrum.
89 L – Absolutely, positively not applicable.
90 M – Can be structured to measure reactivity worth, but not as straightforward as fresh fuel criticals.
91 L – Configuration may not adequately represent reality.
92 H – Strong potential for a good mockup of a cask.
93 M – Same as high, but the uncertainty is higher than other methods. Not as straightforward as fresh fuel critical.
94 L – Same as for cask structural and absorber materials
95 M – Reactor leakage is closer to that of a cask than to that of most lab criticals. Some criticals will have a different spectrum.
96 L – Reactivity magnitude is not sufficient in a reactor core due to size and not due to reflector, so reflector has a smaller influence. This will vary with cask size. The geometry and materials for reflector may be different.
97 L – Absolutely, positively not applicable.
98 Yes – Non-uniformities in a spent fuel system will result in effects that need to be tested.
99 No – Not important enough – covered under other issues ranked above.

100 M – Clean fuel issues indicate areas that should be investigated further in a spent fuel domain. Some issues are adequately validated with fresh fuel criticals.

101 L – Fresh fuel critical cannot represent the effects that are due to spent fuel compositions.

102 L – Absolutely, positively not applicable.

103 L – Absolutely, positively not applicable.

104 M – Last best hope for representing spent fuel phenomena outside a reactor.

105 L – Detector problems are expected in a spent fuel environment (radioactivity problems). Difficulty in discerning effects.

106 H – Interactions are clearly present in a reactor. Cycle-by-cycle variations may allow tracking of trends.

107 M – Similar to above, but the measurements wouldn't be enough to draw firm conclusions because of differences between cask and reactor.

108 L – Reactor interactions in a reactor are different from those in a cask, and it will be impossible to discern effects.

109 L – Not applicable.

Table C-2. Evaluation of Experiments/Measurements for Code and Data Validation

	Actinides Only						"Full" Burnup Credit (Actinides + Fission Products)					
	Importance for Burnup Credit			Need for Additional Experiments			Importance for Burnup Credit			Need for Additional Experiments		
Experiment Type	**H**	**M**	**L**	**H**	**M**	**L**	**H**	**M**	**L**	**H**	**M**	**L**
Fresh Fuel Criticals	4 [1]	4 [2]	1 [3]	3 [4]	3 [5]	3 [6]	4 [7]	3 [8]	2 [9]	4 [10]	1 [11]	4 [12]
Reactivity Worth Measurements												
- Oscillation type (small worth)	2 [13]	4 [14]	3 [15]	0	4 [16]	4 [17]	1 [18]	6 [19]	2 [20]	1 [21]	8 [22]	0
- Direct measurement (large worth)	1 [23]	7 [24]	1 [25]	0	5 [26]	2 [27]	2 [28]	6 [29]	0	2 [30]	6 [31]	0
- Individual Fission Products							6 [32]	2 [33]	1 [34]	6 [35]	2 [36]	1 [37]
- Integral Fission Products							5 [38]	4 [39]	0	3 [40]	6 [41]	0
Subcritical Experiments	1 [42]	4 [43]	4 [44]	1 [45]	2 [46]	6 [47]	1 [48]	2 [49]	6 [50]	1 [51]	2 [52]	6 [53]
Reactor Criticals												
- Measurements	5 [54]	3 [55]	1 [56]	0	2 [57]	7 [58]	4 [59]	5 [60]	0	0	2 [61]	7 [62]
- Documentation				6 [63]	2 [64]	0				7 [65]	1 [66]	0
Radiochemical Assays	4 [67]	5 [68]	0	6 [69]	3 [70]	0	6 [71]	3 [72]	0	7 [73]	2 [74]	0

Footnotes for Table C-2

1 H – Give significant benchmark data for the major actinides.
2 M – Lack of burnup is a detriment.
3 L – Based on Table C-1 rankings.
4 H – Lacking SNF experiments, actinide experiments that 'look like' spent fuel contents are needed. Also, the existing experiments may not address the special considerations of a cask design.
5 M – Existing experiments should be surveyed for appropriateness. Need additional experiments for higher enrichments. There is a lack of relevant experiments at medium and high burnup.
6 L – There is a sufficient database of existing fresh fuel criticals. Additional criticals needed would be a very small addition to existing data.
7 H – Same as Actinides Only
8 M – Same as Actinides Only; also lack of fission products gives them a lower net value.
9 L – Same as Actinides Only.
10 H – Same as Actinides Only.
11 M – Same as Actinides Only.
12 L – No value in adding more Actinides Only experiments to improve the actinide + fission product safety case.
13 H – Very sensitive to cross sections
14 M – Not sufficient by themselves to validate criticality calculations. Only type of measurement that can validate minor actinides.
15 L – Cannot be used in direct criticality code validation.
16 M – More validation is needed for minor actinides; these actinides will become more important with higher burnup and initial enrichments.
17 L – Sufficient understanding of actinide cross sections as a whole.
18 H – Same as Actinides Only.
19 M – Validation of fission product cross sections; however, not a direct measurement that can be used to validate the criticality codes.
20 L – Same as Actinides Only.
21 H – Resolve errors/questions with respect to fission product cross sections.
22 M – More validation is needed for fission products, both individually and integrally.
23 H – French are making progress using this type of experiment (HTC) for BUC.
24 M – Experiments address cross sections, and are 'easily' done, although they don't provide an assessment of a safety margin.
25 L – More useful than oscillator, but there are better experiment types for gathering information.
26 M – Some cask modeling issues can be mocked up in this type of experiment in a straightforward manner.
27 L – Same as for oscillator experiments.
28 H – Same as Actinides Only, plus Valduc fission product experiments. Can test reactivity worth of each individual fission product.
29 M – Same as Actinides Only.
30 H – Can test reactivity worth of each individual fission product. Is limited to reduced set (7) of fission products since some of the lower-worth fission products are not measurable.
31 M – Same as Actinides Only.
32 H – Crucial for validation of individual fission products. Existing fission product cross section evaluations are weaker than those of actinides.
33 M – Same as low ranking, but not as strong.
34 L – Individual worths are quite small – difficult to measure with larger uncertainties.

35 H – Limited access to existing experiments, and a scarcity of such experiments. Needed to validate the fission product production in higher-burnup fuel. Fission product worth increases with increasing burnup. Also can decrease the uncertainty in existing fission product measurements. However, this will impact a smaller subset (7-14) of nuclides.
36 M – Same as High, but does not identify resonance overlap effects.
37 L – No value added for licensing applications.
38 H – Crucial for validation of total fission products.
39 M – Cross-section overlap may be problematic with respect to the safety basis. However, net effect is easier to measure than the individual small worths.
40 H – Provides information on smaller-worth fission products. Also captures resonance overlap effects.
41 M – Provides information on resonance overlap not available elsewhere.
42 H – Very sensitive to cross sections.
43 M – One can approximate a cask and get direct measurements, but with a larger uncertainty than criticals.
44 L – Information is not readily usable in validation process.
45 H – An alternate to fresh fuel criticals.
46 M – Could be done on actual casks and would like to see such a measurement. Also validates ability to predict subcriticality. However, the task is difficult with larger uncertainty than critical experiments.
47 L – Reactivity worth measurements would better serve to meet additional needs. Sufficient set of general experiments already exist.
48 H – Same as Actinides Only.
49 M – Practical limitations of subcritical methods; however, would be a tremendous asset if difficulties are overcome.
50 L – Practical difficulties (above) coupled with larger uncertainties.
51 H – Would provide a full inventory of spent fuel measurement in a cask-like geometry.
52 M – Same as for High, but diluted by increased uncertainty in approach.
53 L – Higher uncertainties, difficult to measure in a high radiation field, difficulty in translating to a subcritical margin, and oscillation measurement are preferred for additional information.
54 H – Actual fuel, identical in many ways to contents of cask.
~~55 M – Reactor criticals are a source of confidence on burnup calculations.~~
56 L – Actinide and fission product concentrations are not well quantified.
57 M – Measurements are going to be needed for advanced reactor/fuel designs and higher burnup. Perhaps it is necessary to request startup with a particular core configuration
58 L – Critical startup testing is going to continue to occur for each new core/reload.
59 H – same as Actinides Only; only full inventory benchmark.
60 M – Cross section overlap may be problematic with respect to the safety basis. The integral effect is valuable but not sufficient.
61 M – Same as Actinides Only
62 L – Same as Actinides Only
63 H – Assembled data is needed by cask designers and users in a single location.
64 M – There is a significant number already "in the pipeline."
65 H – Same as Actinides Only.
66 M – Same as Actinides Only.
67 H – Direct assessment of spent fuel composition.
68 M – Uncertainty due to separation and measurement processes; lack of information on local (pellet) vs. average (core) concentrations.
69 H – Additional need for higher enrichments and burnups, also minor actinides.

70 M - Sensitivity analyses indicate that new data is needed only for confirmation of trends predicted. Such data will soon be available.
71 H - More important than actinides because fission product data is sparse.
72 M - Same as Actinides Only.
73 H - Same as Actinides Only, with addition of fission products. Fission products are even more heavily weighted due to the lack of assay data (relative to actinide data).
74 M - Same as Actinides Only. Not clear that more data will increase the confidence in fission product prediction, especially in light of cost of assay measurements.

Appendix D

Members of the Burnup Credit PIRT Panel

George H. Bidinger

George H. Bidinger is a consulting engineer with nearly forty-two years of experience in nuclear criticality safety outside reactors. After receiving an MS in physics from John Carroll University, he worked in the nuclear industry for six years before joining the AEC/NRC for thirty years as a safety reviewer, a safety inspector, a regulatory requirements specialist, and a group leader. For the past six years as an independent consultant, he has performed criticality safety evaluations, conducted criticality safety assessments, conducted technical reviews, and participated in University of New Mexico sponsored criticality safety training to NRC, DOE, DOE contractors, NRC licensees, and industry representatives. He has coauthored several criticality safety reports and ANS papers. He has been active in the ANS Nuclear Criticality Safety Division, having served in several executive positions, including Program and Division Chair. He has been a member of several working groups writing ANSI/ANS-8 criticality safety standards and has been a member of the ANS Consensus Committee N16, Nuclear Criticality Safety since 1981.

Brent E. Boyack

Brent E. Boyack is one of the two facilitators for the Burnup Credit PIRT Panel. He is a registered professional engineer. He obtained his BS and MS in Mechanical Engineering from Brigham Young University. He obtained his PhD in Mechanical Engineering from Arizona State University in 1969. Dr. Boyack has been on the staff of the Los Alamos National Laboratory for 20 years; he is currently the leader of the software development team, continuing the development, validation and application of the Transient Reactor Analysis Code (TRAC). Dr. Boyack has over 30 years experience in the nuclear field. He has been extensively engaged in accident analysis efforts, including design basis and severe accident analyses of light water, gas-cooled, and heavy-water reactors, reactor safety code assessments and applications, safety assessments, preparation of safety analysis reports, and independent safety reviews. He chaired the MELCOR and CONTAIN independent peer reviews and was a member of the Code Scaling, Applicability and Uncertainty or CSAU technical program group. He has participated in numerous PIRT panels. He has over 70 journal and conference publications and is an active member of the American Nuclear Society.

Richard J. Cacciapouti

Richard J. Cacciapouti from Duke Engineering & Services obtained his BS in Physics from Lowell Technological Institute and a MS in Nuclear Engineering from the University of Lowell. He has over 37 years of experience in nuclear engineering and reactor physics in both managerial and contributory roles for the support of pressurized water and boiling water reactors (PWRs and BWRs). He has supervised or contributed to 45 cycles of reload core physics analyses for the Yankee Rowe, Maine Yankee, Connecticut Yankee, Vermont Yankee and Seabrook nuclear power plants located in New England. He was instrumental in developing the reactor physics reload analysis methods at Yankee Atomic Electric Company. In the burnup credit area, he has been involved in the development of a PWR axial profile database and in the determination of the accuracy of PWR spent fuel burnup records. Mr. Cacciapouti has provided consulting in the area of reactor physics and analysis of incore data on an international level, and has also authored a number of papers on the application of reactor physics methods to operating power reactors.

José M. Conde

José Conde obtained his BS in Physics from the Complutense University of Madrid (Spain). He has been working in the nuclear field since 1984, when he joined the Consejo de Seguridad Nuclear (CSN), the nuclear safety and radiation protection body of Spain. He worked originally in the field of nuclear systems safety and plant initial start-up inspection. For the last 13 years, his responsibilities have lied in the field of fuel behavior and core analysis, including transient and accident analysis and criticality safety. In the latter field, he has participated in evaluations and analysis related with licensing of fresh and spent fuel storage and transport. He is currently in charge of the Nuclear Engineering Division at the CSN. He is participating as a consultant on burnup credit for the IAEA since 1997, and has been active on the expert meetings on burnup credit of the OECD-NEA since 1995. He is currently Chairman of the Technical Advisory Group of the CABRI Water Loop Experimental Program, and represents the CSN at different international groups working in the field of fuel behavior.

David J. Diamond

David Diamond is one of the two facilitators for the Burnup Credit PIRT Panel. He is Head of the Nuclear Energy and Infrastructure Division at Brookhaven National Laboratory where his management responsibilities are for diverse projects dealing with the safety of nuclear energy, including structural analysis, human factors, operational safety, and life extension. His technical expertise is in neutronics as it pertains to nuclear safety issues. He is a consultant to the NRC dealing with issues such as fuel performance and probabilistic/deterministic analysis of reactivity accidents. His consulting work has included work for the Canadian regulatory authority on CANDU reactor physics/safety issues, work for the training simulator industry developing neutron kinetics models, and numerous assignments for the IAEA. He has a PhD in Nuclear Engineering from M.I.T. and is a Fellow of the American Nuclear Society.

Peter Grimm

Peter Grimm obtained his diploma in physics from the Swiss Federal Institute of Technology in Zurich in 1978. He has been working at Paul Scherrer Institute (PSI), Villigen, Switzerland (until 1987 Swiss Federal Institute for Reactor Research, EIR) since 1979 in the field of analytical Light Water Reactor (LWR) neutronics and criticality safety. He is currently task manager for the analysis part of the LWR-PROTEUS experimental project. His activities have included development and validation of LWR static neutronics codes, preparation of neutronics data for transient analysis, fuel cycle studies, as well as design and analysis of integral experiments. He has been involved in criticality safety as part of his work, particularly in criticality calculations for storage racks and internal criticality reviews for PSI nuclear facilities, and has participated in many international benchmark exercises in this field.

Hae Ryong Hwang

Hae Ryong Hwang from Korea Power Engineering Company (KOPEC) obtained his BS in Nuclear Engineering from Seoul National University in South Korea. He obtained his MS and PhD in Nuclear Engineering from Purdue University. He worked at the Korea Atomic Energy Research Institute (KAERI) in the area of fuel integrity, reactor physics and criticality safety for 12 years before joining KOPEC for five years as a Group Manager. For the past 14 years, he has been responsible for the design and engineering of reactor physics and criticality safety analysis for as many as 10 Korean nuclear power plants. He also served as a leader for several R&D projects in vessel integrity, criticality safety and source term. He is a member of the Korean Nuclear Society and an officer of the Korean Association for Radiation Protection.

Raymond L. Murray

Raymond Murray is Professor Emeritus of Nuclear Engineering at North Carolina State University. He received the BS in Science Education and the MA in Physics from the University of Nebraska, studied at the University of California, and obtained the Ph.D. in Physics from the University of Tennessee in 1950. In World War II he was in research and production on the electromagnetic separation of uranium isotopes for the atomic bomb. Later at Oak Ridge he headed the criticality prevention group. From 1950 to 1980 he served as professor at North Carolina State University, helping establish the first university nuclear engineering program and the first university reactor. He was head of the Nuclear Engineering Department from 1963 to 1974. He has been a member of Duke Power's Safety Review Committee and a member of INPO's Advisory Council. In the 1980s he was criticality consultant for Bechtel in the Three Mile Island recovery program. From 1987 to 1993 he served as member and chairman of the North Carolina Low Level Radioactive Waste Management Authority. Dr. Murray is the author of many technical papers and books on nuclear engineering, physics, reactor theory, radioactive wastes, and nuclear energy.

Jens-Christian Neuber

Jens-Christian Neuber has a BS in Physics from the Ruhr University of Bochum, Germany. After his studies he was self-employed working mainly in the fields of probabilistic safety and risk analysis of nuclear fuel processing and reprocessing facilities as well as propagation of pollutants with off-gas, waist air and water. From 1985 to 1989 Mr. Neuber worked in radiation protection and criticality safety assessment for the NUKEM GmbH, Germany. In 1989 he joined Siemens AG (now Siemens Nuclear Power GmbH), Germany, where he is head of the criticality safety analysis group. Mr. Neuber is a member of the Criticality Safety Committee of the German Society of Standardization (DIN - Deutsches Institut für Normung), and he is participating in the consultancies on burn-up credit held by the IAEA once a year since 1997. Since 1998 Mr. Neuber is active in the OECD NEA WPNCS expert group meetings on burn-up credit. He has described his experience in burn-up credit criticality safety analysis of PWR and BWR UOX and MOX fuel arrangements in several publications.

Michaele Brady Raap

Michaele (Mikey) Brady Raap is the chairman of the international Expert Group on Burnup Credit that functions under the auspices of the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA). She obtained her BS, MS and PhD in Nuclear Engineering from Texas A&M University. Dr. Brady Raap has over 15 years experience in nuclear engineering and engineering management. She is currently with Battelle Northwest Division at the Pacific Northwest National Laboratory as the Technical Project Lead for nuclear criticality safety evaluations supporting the design of the Pit Disassembly and Conversion Facility (part of a system to convert weapons materials into commercial-grade fuel). Dr. Brady Raap has strong technical and programmatic expertise in nuclear criticality safety and spent fuel characterization as well as general systems analysis and performance assessment. Prior to joining Battelle in 1999, her experience includes criticality safety engineering at Oak Ridge National Laboratory and at Sandia National

Laboratories (SNL) supporting the programs of the DOE Office of Civilian Waste Management. These programs addressed criticality safety in the storage and transportation of nuclear materials with primary focus on burn-up credit issue for commercial spent fuel. She also served as the lead manager for SNL activities supporting the Yucca Mountain Site Characterization Project, including geo-mechanical monitoring and characterization, systems engineering support and performance assessment.

Joseph Sapyta

Joseph Sapyta has technical, technical management, project management and business development experience. He has more than 36 years experience in the nuclear industry, including nuclear safety and safeguards. His experience spans the entire fuel cycle from fuel enrichment processes to final disposition at the proposed geologic repository. Dr. Sapyta has managed numerous projects for testing, fabrication, licensing, engineering, field service, and disposition of nuclear core components used in commercial nuclear power plants. In addition, he has 10 years experience on various Department of Energy/Department of Defense projects involving extensive work with the National Laboratories. These projects include plutonium disposition, high burnup fuel, nuclear rocket development, fuel cell demonstrations, and the Yucca Mountain Project. Dr. Sapyta has participated on numerous working groups involving these projects. Currently, Dr. Sapyta is a consulting engineer with the Waste Package Design Department on the Yucca Mountain Project. He is a co-author of the *Disposal Criticality Analysis Methodology Topical Report* and has participated in the associated presentations to the Nuclear Regulatory Commission.

Daniel Thomas

Daniel Thomas has a BS and MS in Nuclear Engineering from the Georgia Institute of Technology, Atlanta, Georgia. He is the criticality department manager for the Civilian Radioactivity Waste Management System Managing and Operating Contractor on the U.S. Department of Energy's Yucca Mountain Project in the area of criticality since 1993, first for Framatome ANP and most recently for Bechtel SAIC Company, LLC. His major involvement in the burnup credit activities have included supporting the development of the Office of Civilian Radioactive Waste Management's Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages, being the lead author of the Yucca Mountain Project's Disposal Criticality Analysis Methodology Topical Report which includes consideration of burnup credit for disposal, and presenting at the IAEA Technical Committee Meetings on Burnup Credit.

Robert E. Wilson

Robert Wilson has a BS and MS in Engineering Physics from the University of California, Los Angeles and a PhD in Nuclear Engineering, from the University of Washington. His dissertation research topic was Critical Mass Physics, and his post-doctoral research topic was Fast Reactor Safety. He is a Fellow of the American Nuclear Society and is the 2001-2002 Chairman of the Nuclear Criticality Safety Division of the society. Dr. Wilson was the Criticality Safety Program Manager for the DOE Rocky Flats Field Office. He was the Criticality Safety Manager at the Idaho Chemical Processing Plant and the Rocky Flats Closure Site. He served as a Criticality Specialist for the Nuclear Regulatory Commission.

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Richard Y. Lee, Project Manager

11. ABSTRACT *(200 words or less)*

When fuel is irradiated in a reactor, the isotopic composition, and therefore the reactivity, of the fuel changes with burnup. Nevertheless, until recently, the NRC's approval of the criticality safety evaluations for commercial spent fuel in casks had been based on analyzing the spent fuel as though it were unirradiated. This "fresh fuel" assumption provided a well-defined, straightforward, and bounding approach that eliminated the need for information on the reactor operating conditions and reactor power history and the core history of individual fuel assemblies. However, while being bounding, the fresh fuel assumption leads to overly conservative cask designs. To address this conservatism, the NRC issued the Spent Fuel Project Office Interim Staff Guidance - 8 Rev.1 based on information available at the time. ISG8R1 allows limited burnup credit (BUC) in the criticality safety analyses of Pressurized Water Reactor (PWR) spent fuel in transport and storage casks. In the same time frame of ISG8R1, the NRC initiated a research program to: clarify guidance on acceptable technical approaches to BUC; develop approaches for expanding the range of BUC; and address regulatory needs for safe, simple, and cost-effective implementation of BUC. As an element in this research program, the NRC commissioned the formation of a Phenomena Identification and Ranking Table (PIRT) panel. Two objectives of the PIRT panel were to 1) identify and rank phenomena, processes, and parameters, that influence the effective neutron multiplication factor (k_{eff}) for irradiated nuclear fuel in transport and storage configurations, and 2) assess the worth of experimental measurement facility classes for validation of calculational methods. Results of the panel's findings are presented in this report.

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