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Review of Information for Spent Nuclear Fuel Burnup Confirmation

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Manuscript Completed: June 2009
Date Published: December 2009

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ABSTRACT

The Interim Staff Guidance on burnup credit (ISG-8, revision 2) for pressurized-water-reactor spent nuclear fuel in storage and transport casks, issued in 2002 by the U.S. Nuclear Regulatory Commission's Spent Fuel Project Office, recommends an out-of-core burnup measurement to confirm the reactor record and compliance with the assembly burnup value used for cask loading acceptance. This recommendation is intended to prevent unauthorized loading (i.e., misloading) of assemblies due to inaccuracies in reactor burnup records and/or improper assembly identification, thereby ensuring that the appropriate subcritical margin is maintained. The purpose of this report is to detail information and issues relevant to preshipment burnup measurements when using burnup credit in pressurized-water-reactor spent nuclear fuel transport and storage casks. In particular, this report reviews the role of burnup measurements in the regulatory guidance for demonstrating compliance with burnup loading criteria, burnup measurement capabilities and experience, generation and accuracy of utility burnup records, fuel movement and misloading experience, and the consequences of misloading assemblies in casks designed for burnup credit.

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EXECUTIVE SUMMARY

This report is only applicable to pressurized-water-reactor (PWR) spent nuclear fuel (SNF). No information was evaluated for boiling water reactors.

Criticality safety analyses for commercial PWR spent fuel storage and transport have historically made the conservative assumption that the SNF is fresh (unirradiated), with uniform isotopic composition corresponding to the maximum allowable enrichment. Hence, the criticality-safety-related loading criteria are based on as-manufactured fuel design specifications. While this “fresh fuel” assumption provides a conservative, well-defined approach to the criticality safety analysis that is not dependent on the fuel operating history, it ignores the decrease in reactivity that occurs as a result of irradiation. Taking credit for the decrease in reactivity due to irradiation is termed burnup credit. In contrast to safety analyses that use the fresh fuel assumption, burnup credit analyses necessitate consideration of the fuel operating history, additional validation of calculational methods (due to prediction and use of SNF nuclide compositions), consideration of new conditions or configurations for the licensing basis, and additional measures to ensure proper cask loading. A “burnup credit cask” refers to a cask where burnup credit is implemented in the design and safety analysis. A burnup credit cask can enable an extended range of allowable contents and/or increased cask capacities but requires knowledge of the SNF characteristics commensurate with the criteria (e.g., minimum discharge burnup) for approved contents as established in the safety analysis.

The current Nuclear Regulatory Commission (NRC) guidance for use of burnup credit in transport and storage casks is limited to PWR SNF and was issued in 2002 as interim staff guidance by the NRC Division of Spent Fuel Storage and Transportation (formerly the Spent Fuel Project Office). This interim staff guidance (referred to as ISG-8, rev. 2) includes a recommendation for a measurement to confirm knowledge of a key PWR SNF characteristic related to reactivity (e.g., discharge burnup) and help prevent loading of a more reactive assembly than that permitted for the approved burnup credit cask contents. A burnup measurement is recommended, in part, to verify the accuracy and completeness of utility reactor burnup records for all SNF and allay concerns associated with the potential for calculational, procedural, and other human errors in selection of assemblies for loading.

The purpose of this report is to detail information and issues relevant to preshipment burnup measurements when using burnup credit in PWR SNF storage and transport casks. In particular, this report reviews the role of burnup measurements in the regulatory guidance for demonstrating compliance with burnup loading criteria, burnup measurement capabilities and experience, generation and accuracy of utility burnup records, fuel movement and misloading experience, and the consequences of misloading assemblies in casks designed for burnup credit.

Over the past 20 years, there have been multiple SNF out-of-core examinations conducted at U.S. utilities to compare utility records for assembly burnup and cooling time with out-of-core measurement data. The measurement equipment used [neutron/gamma detection systems (e.g. the “Fork Detector” system from Los Alamos National Laboratory) and high-resolution gamma-ray spectrometry (HRGS) systems available commercially from several vendors] was developed to detect fissile material diversion and was not specifically designed for ease of use in reactor spent fuel pools or optimized to accurately verify or quantify SNF burnup independent of utility reactor records. However, the data gathered from these examination campaigns have been fairly uniform and consistent and are useful for evaluating the effectiveness of available measurement techniques.

The following observations, also provided in Section 8, highlight key points from the information reviewed for this report.

Reactor Burnup Records

- Utility records for fuel burnup are based on either (1) the measured core thermal output, with burnup distributed to individual assemblies using validated computer codes, or (2) a combination of information provided by in-core detectors, measured core thermal output, and validated computer codes.

- A significant amount of data from 1980 to the present is available to support a finding that utility records for fuel burnup are accurate for individual spent fuel assemblies to at least 5% of "true" assembly burnup. These data primarily originate from previous ex-core burnup measurement programs, comparisons between calculated burnup values (on which reactor record values are based) and burnup values inferred from in-core measurements, and retrospective evaluations based on comparisons between existing reactor record values and calculated values.

- Utilities do not all use the same methods to calculate and verify the assembly burnup values that are recorded in their reactor records. Moreover, the computational methods used by many utilities have evolved over time such that burnup values for older and newer fuel are based on different methods. Hence, for SNF assemblies to be transported in burnup credit casks, utilities need to demonstrate how the burnup values in their reactor records were developed and recorded and document information as to the accuracy of their recorded burnup values. Examples of this type of activity are the reactor record verification programs described herein by Duke Energy and the Tennessee Valley Authority (TVA).

- In at least one case, burnup values in a utility's reactor records for some assemblies are based on the "batch average" burnup, which is definitely not an accurate representation of the individual burnup for each of the assemblies. The number of utilities that may have used batch average data is unknown. However, based on a review of various data sources for discharged fuel, it is expected that the use of batch averaged values is limited in number and would apply to older fuel records. Contemporary reactor records use individual, assembly-specific burnup calculations and values.

- Unless a quality assurance program is implemented to ensure the accuracy of the reactor records, reactor records may become less accurate over time due to transcription errors as record media degrade or are changed to newer media or as software is updated.

Burnup Verification Measurements

- Out-of-core measurement systems can adequately verify reactor record burnup information. However, as is also the case for in-core measurement systems, out-of-core measurement systems cannot measure fuel burnup directly. Instead, these out-of-core measurement systems measure gamma-ray and/or neutron emissions from the assemblies, which are then compared to a calibration curve to develop an estimated fissile content and corresponding assembly burnup.

- The burnup measurement programs reviewed for this report concluded that out-of-core measurement systems provided somewhat less accurate burnup values, as compared to reactor record burnup values developed using reactor in-core monitoring systems and design codes. The modern in-core systems, which use core measurements and some data from design codes, produce burnup values that are generally within approximately 2% of the burnup values predicted by design codes, whereas the out-of-core measurement systems produce average assembly burnup results that are within the expected fork detector accuracy range of 2.2–5% of "true" burnup. However, there were some fork detector examinations with maximum assembly deviations as high as 9.1%. Regardless, these examination programs demonstrated that out-of-core measurements could identify substantially underburned fuel assemblies and hence could be used to prevent them from being accidentally loaded into spent fuel transport casks.

- Careful calibration against an assembly of known burnup, known cooling time, and identical geometry is required to achieve the reported accuracies with the fork detector used in the U.S. studies. Thus, in practice, these out-of-core measurements are dependent on and calibrated against reactor burnup records. Consequently, the burnup values inferred from these measurements tend to have a higher uncertainty than reactor record assembly burnup values determined using in-core measurements.

- Fuel assembly axial-burnup profiles have a significant impact on reactivity and are therefore an important component in determining "average assembly burnup." For the fork detector examination programs evaluated in this report, personnel used measurements at the assembly centerline (midplane) and assumed that the axial profile of the reference assembly could be used to estimate the assembly average burnup (i.e., it was assumed that the axial profiles were the same for the reference and measured assemblies). This approach could give erroneous burnup measurement results for assemblies that have different axial burnup characteristics. Hence, if out-of-core measurements are used, care should be taken to ensure that correct average burnup information is collected, commensurate with the measurement accuracy goals and criteria for approved contents.

- The costs and risks associated with performing out-of-core burnup measurements should be balanced with the risks and potential costs of not performing the measurements. Out-of-core measurement campaigns require utility resources for planning and execution, increase the dose to personnel, increase the risk of damage to assemblies and potential fuel mishandling events due to the increased assembly movements, and have associated financial cost to the utility. The risks and potential costs associated with loading a significantly underburned assembly into a transport or storage configuration have not been explored in this report.

- The neutron-counting measurement systems appear to be more accurate than the HRGS systems and require less skilled operators for handling the SNF assemblies.

- Considerable efforts, primarily motivated by interests related to nuclear material safeguards, are ongoing to develop better and more accurate measurement systems.

Consequences of Fuel Assembly Misloading

- The consequences to k_{eff} of loading assemblies that have slightly reduced burnup (e.g., 5%) due to uncertainties in the burnup verification process), as compared with the required burnup, are fairly small ($\leq$ 1%). On the other hand, loading one or more highly enriched (i.e., >4 wt %) fresh fuel assemblies has a significant consequence on criticality safety. These findings suggest that while it may not be overly important to precisely verify the burnup value, it is important to ensure that fresh or very-low-burnup (nearly fresh) fuel assemblies are not misloaded into a burnup credit cask.

Fuel Movement and Operational Considerations

- Although utilities' record of reliability in selecting and moving assemblies during fuel-handling processes may be characterized as "good" from the standpoint that no inadvertent criticalities have occurred as a result of fuel misloading, there are a number of documented examples of fuel misloading. Considering the large number of fuel assembly movements that have been executed, relatively few mishandling events involving movement of an incorrectly identified fuel assembly have been reported. Procedural violations related to SNF movements that resulted in violations of plant Technical Specifications have occurred with a fairly low frequency. Most of these events were not a result of incorrect burnup values assigned to the SNF but were instead the result of personnel error in selecting assemblies for movement.

- Fresh fuel is visibly different from SNF due to the oxidation, crud buildup, and bending/twisting of the latter. Visual inspection should easily differentiate new assemblies from SNF. There is some uncertainty about the appearance of an assembly with very limited burnup (e.g., removed promptly after start-up due to leaking fuel rods or some other problem) after it has resided in a spent fuel pool for a number of years.

- Visual inspections and/or simple field measurements (e.g., gross radiation measurements, Cerenkov radiation detector) could be performed during cask loading to detect and prevent accidental loading of fresh or nearly fresh fuel. A basic detector system could be devised to ensure that an assembly has some minimum activity level. One examination tool that may suffice is the digital Cerenkov viewing device (DCVD). Any measurement program would have to be evaluated to determine its limitations to detect SNF. For example, the DCVDs may not be able to differentiate between very old, moderately burned SNF and very-lightly-burned (i.e., nearly fresh) SNF. Also, any examination limitations, such as equipment alignment for Cerenkov examinations, must be evaluated to ensure that adequate examination results are achievable for all SNF to be evaluated.

ABBREVIATIONS

ALARA	as low as reasonably achievable
ANO	Arkansas Nuclear One
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARMP	Advanced Recycle Methodology Program
B&W	Babcock & Wilcox
BWR	boiling water reactor
CoC	Certificate of Compliance
CVD	Cerenkov viewing device
DCVD	digital Cerenkov viewing device
EPRI	Electric Power Research Institute
GWd/MTU	gigawatt-days per metric ton of uranium initially loaded into an assembly
HRGS	high-resolution gamma-ray spectroscopy
IAEA	International Atomic Energy Agency
ID	identification (number)
k_{eff}	effective neutron multiplication factor
LER	Licensee Event Report
LWR	light water reactor
NCTL	Nuclear Component Transfer List
NRC	Nuclear Regulatory Commission
pcm	percent mille, which means thousandths of a percent
	pcm = 1×10^{-5} $\Delta k/k$
PWR	pressurized water reactor
QA	quality assurance
RE	residual enrichment

SCSS	Sequence Coding and Search System (NRC)
SDM	shutdown margin, as in reactor shutdown margin
SNF	spent nuclear fuel
SNM	special nuclear material
TMI	Three Mile Island
TVA	Tennessee Valley Authority
^{235}U	uranium isotope with 143 neutrons
UV	ultraviolet
ZPPT	zero power physics testing

1 INTRODUCTION

This report is only applicable to pressurized-water-reactor (PWR) spent nuclear fuel (SNF). No information was evaluated for boiling water reactors (BWRs).

Criticality safety analyses for commercial PWR spent fuel storage and transport have historically made the conservative assumption that the SNF is fresh (unirradiated), with uniform isotopic composition corresponding to the maximum allowable enrichment.[1] Hence, the criticality-safety-related loading criteria are based on as-manufactured fuel design specifications. While this "fresh fuel" assumption provides a conservative, well-defined approach to the criticality safety analysis that is not dependent on the fuel operating history, it ignores the decrease in reactivity that occurs as a result of irradiation. Taking credit for the decrease in reactivity due to irradiation is termed burnup credit. In contrast to safety analyses that use the fresh fuel assumption, burnup credit analyses necessitate consideration of the fuel operating history, additional validation of calculational methods (due to prediction and use of SNF nuclide compositions), consideration of new conditions or configurations for the licensing basis, and additional measures to ensure proper cask loading. A "burnup credit cask" refers to a cask where burnup credit is implemented in the design and safety analysis. A burnup credit cask can enable an extended range of allowable contents and/or increased cask capacities but requires knowledge of the SNF characteristics commensurate with the criteria (e.g., minimum discharge burnup) for approved contents as established in the safety analysis.

The reduction in SNF reactivity due to irradiation is addressed in the U.S. Nuclear Regulatory Commission (NRC) *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, NUREG-1617[2] and *Interim Staff Guidance–8, revision 2, Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks* (ISG-8, rev. 2),[3] which provides recommendations for the acceptance of a burnup credit approach in the criticality safety analysis of PWR spent fuel casks. The products of a burnup credit safety evaluation typically include loading curves, which specify cask-loading criteria in terms of the minimum required burnup as a function of initial assembly enrichment. A loading curve represents combinations of burnup and initial enrichment that correspond to a limiting value of the effective neutron multiplication factor (k_{eff}) for a given configuration (e.g., a cask). Figure 1.1 is an example of a loading curve. Assemblies with insufficient burnup, as compared with the loading curve, are not acceptable for loading. Misloading of an underburned (i.e., relative to the loading curve) fuel assembly into a cask will cause an increase in total reactivity. The extent of the reactivity increase is dependent on several factors but is dominated by the amount by which the actual assembly burnup is less than the minimum burnup value for loading acceptance and the position of the assembly in the cask. It is worth noting at this point that the number or percentage of assemblies that do not meet the burnup requirement for loading, and hence have the potential to be misloaded, is dependent on the location of the loading curve relative to the discharged SNF inventory, ranging from a fairly large percentage for loading curves corresponding to actinide-only burnup credit and high-capacity casks (see for example Fig. 1.1) to a fairly small percentage or even zero for loading curves corresponding to full (i.e., actinides and fission products) burnup credit and/or low capacity casks.

Recognizing the importance of assembly burnup as a loading criterion with burnup credit, NUREG-1617 and ISG-8 both include a section to address the issue titled "Assigned Burnup Loading Value." The following excerpt from ISG-8, rev. 2, which was released in September 2002, provides the most recent guidance available.

> Administrative procedures should be established to ensure that the cask will be loaded with fuel that is within the specifications of the approved contents. The administrative procedures should include a measurement that confirms the reactor record for each assembly. Procedures that confirm the reactor records using measurement of a sampling of the fuel assemblies will be

considered if a database of measured data is provided to justify the adequacy of the procedure in comparison to procedures that measures each assembly.

The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For confirmation of assembly reactor burnup record(s), the measurement should provide agreement within a 95% confidence interval based on the measurement uncertainty. The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by a combination of the uncertainties in the record value and the measurement.[3]

The recommendation for a measurement (generally accepted to be an out-of-core measurement) is included to ensure that a burnup credit cask is not incorrectly loaded with a more reactive assembly (including a fresh fuel assembly) than that permitted by the loading criteria.

The purpose of this report is to detail relevant information and issues related to preshipment burnup measurements when using burnup credit in PWR SNF transport and storage casks. In particular, this report reviews the role of burnup measurements in the regulatory guidance for demonstrating compliance with burnup loading criteria, burnup measurement capabilities and experience, generation and accuracy of utility burnup records, the consequences of misloading assemblies in casks designed for burnup credit, and fuel movement and misloading experience.

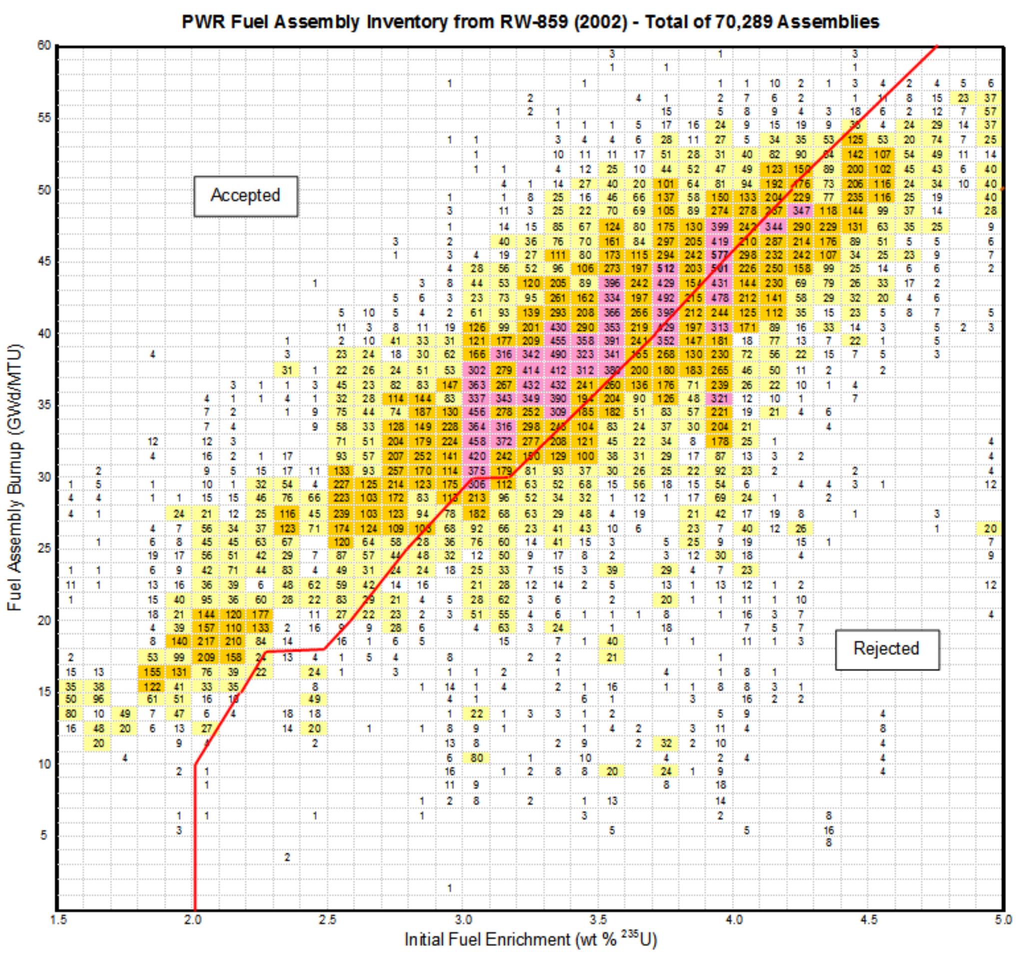

Fig. 1.1. Illustrative loading curve superimposed on the U.S. discharged pressurized-water-reactor (PWR) fuel inventory.

2 ROLE OF BURNUP MEASUREMENTS IN DEMONSTRATING COMPLIANCE WITH BURNUP LOADING CRITERIA

A Certificate of Compliance (CoC) for a spent fuel transportation cask contains technical requirements and operating conditions (fuel specifications, cask leak testing, surveillance, and other requirements) for the cask design and specifies the authorized contents for the cask system. Fuel specifications for authorized contents that are influenced by criticality safety considerations include fuel assembly design and initial fuel enrichment. Uncertainties and/or tolerances in the assembly design parameters and initial fuel enrichment are considered and, where appropriate, accounted for in the criticality safety evaluation. The specifications listed in the CoC are all verified before assembly loading, typically via comparison with utility records for the assemblies. With burnup credit, fuel burnup (and cooling time) becomes a parameter that must be included in the CoC, typically in the form of a loading curve (see Fig. 1.1), and verified prior to assembly loading. As with any parameter that affects safety (e.g., initial fuel enrichment), burnup must be verified to ensure compliance with the safety criteria. However, unlike initial fuel enrichment, which has a well-defined, as-manufactured value with a known uncertainty, burnup is a measure of the power production of a fuel assembly during its residence in a reactor core. Hence, the issues for burnup verification are as follows: (1) the assembly burnup value and (2) the uncertainty in the assembly burnup value.

Two means for verifying assembly burnup include (1) comparison with reactor record data, which include assembly burnup from core-follow calculations supplemented by in-core flux measurements (to varying extents) and (2) out-of-core, post-irradiation burnup measurements. Out-of-core burnup measurements provide a "defense-in-depth" backup or verification of the reactor record data to prevent unauthorized loading (i.e., misloading) of assemblies due to inaccuracies in the reactor burnup records and/or improper assembly identification, thereby ensuring that the appropriate subcritical margin is maintained. The burnup measurements are recommended in the current regulatory guidance to verify that SNF assemblies meet the cask loading criteria and confirm the proper fuel assembly selection before loading. Although available out-of-core measurement techniques must be calibrated against reactor burnup records, they can provide a means to quantify the uncertainty in the reactor record data that can subsequently be used to determine an assembly burnup value to be used for loading acceptance (termed the "assigned burnup loading value" in ISG-8). The assigned burnup loading value is the confirmed reactor record value as adjusted by reducing the record value by a combination of the uncertainties in the record value and the measurement. Hence, large uncertainties in the measured and/or reactor burnup records will result in higher values of burnup being required for loading (for a given initial enrichment) and, subsequently, a reduced loading acceptance (as compared with the unadjusted burnup value; see Fig. 1.1).

3 OUT-OF-CORE BURNUP MEASUREMENTS—EXPERIENCE AND CAPABILITIES

3.1 MEASUREMENT TECHNIQUES

As it is not possible to measure assembly fissile content or burnup directly, existing burnup measurement equipment measures neutron and gamma radiation emitted from an assembly and uses this information, along with calibration data, to infer the assembly burnup. For example, some of the transuranic elements formed in nuclear fuel from successive neutron absorptions have relatively short spontaneous fission half-lives and thus emit relatively large numbers of neutrons (n) and/or decay by alpha (α) emission at a sufficient rate to produce significant neutrons from the α-n reaction. The rates of such neutron emissions have a very strong dependence upon burnup. A complicating aspect of such measurements is that, after significant exposure in a reactor, the radiation signatures of interest may be masked by radiation from fission products, activated structural components, and transuranic elements that build up as a result of the fission process.

"Active" neutron detection systems measure the enhanced neutron flux from induced fission events that occur in the residual fissile material. Some active measurement systems use pulsed sources; others use randomly pulsed sources or steady state sources. The sources are typically neutron sources but could also be high-energy gamma-ray sources for gamma-fission reactions. The detectors are used to detect prompt and delayed neutrons and/or gamma rays. The count rate is proportional to the fissile material in some cases, but often the time dependence of the count rate is used to quantify the amount of fissile material. Such measurements require calibration standards. In general, active neutron counting systems are not typically used in spent fuel pools because the background neutrons produced by the ^{244}Cm in the SNF result in relatively long measurement times for convergence of time-dependent detector count rates. The convergence of the time-dependent data is directly related to the source-induced fission rate relative to that resulting from the inherent fission process as a result of ^{244}Cm and other spontaneous fission isotopes that are present in SNF. In research reactors, the "background" neutrons produced by the ^{244}Cm are low enough that several active measurement techniques can be used to determine fissile content in the SNF.

For gamma-ray detection, the events being measured are counts in specific, narrow energy bands that are characteristic of selected individual fission products or groups of fission products. The isotopes selected for measurement ideally should have a high fission yield, a long half-life, and an energetic gamma whose energy falls in a band not shared by other fission products. Because of the short half-life of some of the primary gamma-ray emitters, the fission product, ^{137}Cs, which has a half-life of 30.07 years and a gamma-ray energy of 661.7 keV, is the major gamma-ray emitter measured after 5 years of SNF cooling. Cesium-137 also has the favorable characteristics that its neutron absorption cross sections are negligible and its yields from both ^{235}U and ^{239}Pu are approximately the same. Because the gamma rays of interest are typically only a small fraction of the total incoming gamma rays, the problems of background and background subtraction are significant, requiring longer counting times to obtain adequate statistical accuracy within the very narrow energy bands needed for good resolution.

"Passive" neutron detection systems measure spontaneous fission neutrons from higher actinides, mainly ^{244}Cm. Passive neutron examination methods have a strong count rate dependency on burnup, cooling time, and initial enrichment. Curium-244 has a half-life of 18.1 years and is formed by successive neutron captures, beginning with ^{238}U. After 10 years of cooling, the ^{244}Cm provides ~95% of the source neutrons and by the 20th year of cooling, still provides nearly 92% of the source neutrons.[4] The neutron emission of ^{244}Cm is very sensitive to variations in assembly burnup. Additionally, measurement of the ^{244}Cm neutrons is very sensitive to many physical factors in the measurement setup (such as equipment positioning errors) that could affect the neutron count. Factors such as boron concentration of the spent fuel pool water must also be known and replicated if the equipment calibrations are not performed in conjunction with the SNF measurements. Additionally, because of the relatively short half-life of ^{244}Cm, the cooling time of the fuel must be known to correctly interpret the total neutron output. Operator-declared values for the cooling time must be used. These cooling times can be verified by several measuring techniques, such as measuring the total gamma-ray activity of the assembly.[5]

3.2 PRACTICAL AND OPERATIONAL CONSIDERATIONS

Relevant issues associated with the use of out-of-core measurements for burnup verification include the following: accuracy and reproducibility of the out-of-core measurements and related calculations; accuracy of the reactor records used for equipment calibration; accuracy of the burnup measurement relative to reactor records; potential increase in probability for fuel-handling accidents due to the increase in fuel-handling operations; any additional radiation exposure to personnel; cost of the equipment installation and use; and the impact of the measurement activity on other ongoing utility operations, including both the duration and degree of interference with other operations (almost all out-of-core measurement devices require the use of the operator's bridge crane). The issues of accuracy and reproducibility of the measurements and related calculations, accuracy of reactor records used for measurement calibration, accuracy of the burnup measurement relative to reactor records, and potential increase in probability for fuel-handling accidents due to the increase in fuel-handling operations are discussed in separate sections of this report.

The impacts of the out-of-core measurement activity on other ongoing utility operations and any additional radiation exposure to personnel are important topics and are briefly discussed in this report, but adequate data to develop conclusions were not available. Cost information in the form of potential cost-benefit analyses is available, but the cost information is highly dependent on assumptions that may be easily biased, thus making the analyses very judgmental. For that reason, cost information was not included in this report.

3.3 ISSUES AFFECTING MEASUREMENT EQUIPMENT ACCURACY

The accuracy of any out-of-core burnup measurement device is primarily a function of the equipment's ability to discriminate between the radiation of interest and the background radiation field, the quality of the equipment calibration and subsequent equipment operation, and the ability of the operator to properly use the equipment and interpret the data in terms of burnup. The radiation data must further be interpreted as to what fraction of a fuel assembly it represents and what this fraction implies with respect to the whole assembly.[4] When using radiation-monitoring equipment to verify assembly burnup, certain parameters must be known or assumed. For example, in the ideal case, a measured count rate would be related (via knowledge of detector efficiency) to the total decay rate of the isotope in that part of the assembly and thus (via a known isotope half-life) to the total inventory of the fission product isotope. Then, via knowledge of the percent fission yield of the isotope, the total number of fissions would be known. Via knowledge of the energy yield per fission, the total energy output, and hence the burnup of that part of the assembly, would be determined. The ability to achieve high accuracy in the determination of the burnup from count rate in the ideal situation depends upon knowledge of four physical parameters and three favorable constraints. The physical parameters are detector efficiency, isotope half-life, percentage fission yield, and fission energy yield. The constraints are direct fission yield of the isotope, very low neutron cross section (avoiding in-core depletion of the fission product isotope), and long half-life (avoiding significant decay of the isotope before measurement). In the absence of accurate knowledge of the absolute detector efficiency, a calibration of the counting system can be accomplished if a geometrically similar fuel assembly of well-known burnup is used as a primary standard. In summary, the interpretation of the counting data can be accomplished in the ideal case either through accurate knowledge of absolute detector efficiency and three other physical parameters or by calibration of the system with a primary standard fuel assembly of known burnup and of nearly identical characteristics (except for burnup). The result of the interpretation in this simple ideal case is that the burnup of the measured part of the assembly can be determined directly from the count rate when this value is compared with the count rate and the known burnup of the standard assembly.[4]

Considering the situation that is encountered in actual practice, the three favorable constraints of the ideal case are not encountered, and the four physical parameters of the ideal case are not all known with precision. The actual situation is as follows.

1. The fission product of interest does not have a significantly long half-life. The principal consequence of this is that a single measurement or measurement ratio cannot yield both the burnup and the age; two measurements or measurement ratios of isotopes with different time behavior patterns are necessary if both assembly burnup and age of a spent fuel assembly are to be determined. A second consequence of the time-varying behavior of measured isotopes is that a portion of these isotopes will decay before fuel discharge and the decay fractions will be different for different core residence times and burnup values. Because the cooling age is measured from discharge, corrections for different predischarge decay should be made. The practical effect of such corrections is that postdischarge count rates are no longer proportional to burnup.

2. The fission yield of the isotope is not direct. In fact, many of the isotopes of interest may be first or second daughters of direct-yield fission products or may not even result from fission—they may be the product of one or more serial captures of neutrons. Furthermore, fission yields depend on the fissioning material.

3. Some isotopes of interest may not have small neutron cross sections, and hence, neutron capture will transmute that isotope into other isotopes.

4. Absolute detector efficiency (fraction of total decays that are measured) is very difficult to calculate accurately. Some of the other four physical parameters may be known quite accurately. Many decay half-lives are known with sufficient accuracy that they do not contribute significantly to overall uncertainty. The energy yield per fission (about 200 MeV/fission) is somewhat dependent upon the fissioning element but is generally known quite accurately.

Additionally, any calibration of the counting system will use a geometrically identical fuel assembly with an inexact burnup that is taken from utility reactor records. Thus, the preceding discussion conveys some of the complexity of the processes influencing burnup measurement that occur when fuel is irradiated over extended periods of time (note that total fuel "age" includes irradiation time in the reactor and time after assembly discharge from the reactor). Simple models of measurable events cannot, in most cases, adequately represent the complex and interdependent processes that are taking place. The measurement approach that has evolved is a close coupling of direct calibration with the use of experimentally based analytical models of fuel behavior to interpolate between calibrations or, if necessary, extrapolate beyond calibrations.

Table 3.1 summarizes the advantages and disadvantages of various nondestructive measurement techniques such as passive neutron counting or gamma-ray spectroscopy.[6]

Table 3.1. Advantages and disadvantages of various burnup characterization techniques

Technique	Advantages	Disadvantages
Absolute count rate of the 662-keV gamma ray from ^{137}Cs	• Simple linear relationship between ^{137}Cs and burnup • Half-life of 30 years • Insensitive to variations in reactor power rating and dwell time	• Absolute measurement requires a well-defined and reproducible geometry between the detectors and the fuel assembly
The nuclide activity ratio: $^{134}Cs/^{137}Cs$	• The ratio method makes it insensitive to geometry	• Half-life of ^{134}Cs (2.2 years) requires significant decay correction and can be applied only to fuel with cooling time < 20 years • Burnup correlation is dependent on initial enrichment and power rating
The nuclide activity ratio: $^{106}Ru \times {}^{137}Cs/({}^{134}Cs)^2$	• Insensitive to geometry • Independent of enrichment and power rating	• Useful only for fuel with cooling time < 9 years (^{106}Ru has a 372-day half-life)
Passive neutron measurement (predominantly from ^{244}Cm)	• The neutron signals are received uniformly from all pins in the assembly (gamma-ray measurements are sensitive only to the outer pins) • Good for safeguards applications as it is sensitive to missing or removed fuel pins	• Curium-244 inventory is a strong function of initial enrichment • Neutron assay is very geometry sensitive and can also be affected by multiplication and neutron poisons in the pool or within the assembly

3.4 MEASUREMENT SYSTEMS USED AT COMMERCIAL NUCLEAR PLANTS

Several spent fuel measurement devices are available to support SNF burnup verification. These devices generally use high-resolution gamma-ray spectrometry (HRGS) and passive neutron counting to measure both neutron and gamma-ray emissions from the spent fuel assemblies. When used with depletion computer codes, these measurement devices can determine the SNF assembly burnup and radionuclide inventories.[7,8] The "fork" detector system, developed by Los Alamos National Laboratory to support the detection of potential fissile material diversion from SNF (Fig. 3.1), is designed to determine the extent of the variation among assembly burnup values and to identify any anomalous values.[9] The fork system uses gross gamma-ray ion chambers in addition to the neutron detectors and has been used at several utility spent fuel pools for the independent verification of declared fuel burnup values in connection with International Atomic Energy Agency (IAEA) safeguards inspections. With proper calibration and corrections, the fork detector can determine the burnup of individual fuel assemblies to an average accuracy of about 5% of plant records.[5] The observed data must be corrected for both the variation in cooling times among the assemblies and initial assembly enrichment. The cooling time correction is accomplished by extrapolating the neutron data (after background subtraction) back to the date of discharge of each assembly using an exponential factor with a half-life of 18.1 years, the half-life of the principle neutron emitter, ^{244}Cm. The initial enrichment correction is required because ^{244}Cm, which produces the neutrons, is produced by activation of ^{238}U and is determined by the reactor flux rather than the fission rate. This correction (described in detail in Reference 10) is accomplished by using a factor to adjust the observed count rates for the variation in initial enrichment among the assemblies, using the information in reactor records for initial enrichment.[9] Another similar system, developed by Westinghouse, was demonstrated on PWR fuel at the Surry nuclear power station.[4]

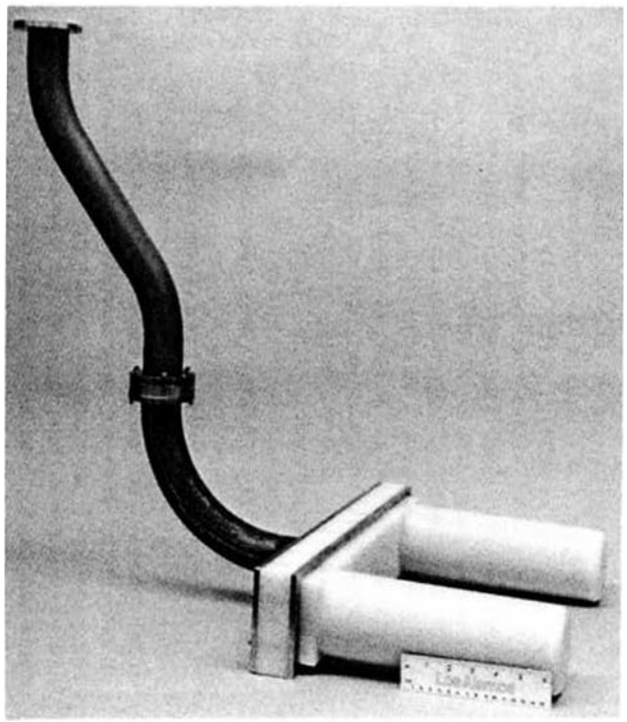

Fig. 3.1. Neutron/gamma-ray detector head, referred to as the "fork." (*Source:* Reference 8.)

3.4.1 Fork Detector Systems

3.4.1.1 Calibration Methodology

Central to the measurement methodology of the out-of-core measurement equipment is use of a reference assembly of known burnup and cooling time and of similar geometry to the assemblies being examined. The use of this reference assembly is essential because the analysis requires the detection and counting of specific events in a fuel assembly of known reference burnup, B_r, and known reference age, t_r. A reference net count rate, C_r, can then be obtained. Subsequently, an assembly of interest can be measured that has nominally identical geometry and initial enrichment but an undefined burnup, B, and age t. The corresponding net count rate is C. An analytical model is then used to characterize the burnup-dependent and time-dependent behavior of the isotope being counted. The basic form of the general-purpose characterization methodology for any specific isotope would then be

$$\left(\frac{C}{C_r}\right) = \left(\frac{B}{B_r}\right)^p \left(\frac{t}{t_r}\right)^{-m} , \qquad (1)$$

where the *p* and *m* coefficients are determined directly from the analytical results for any particular isotope or group of isotopes over the range between the reference burnup and age and the approximate burnup and age of the assembly being measured.[4] Specific values of these coefficients for neutrons, gamma-ray, and heat have been published as part of Appendix 1C of DOE/RW-0184-R1 (Ref. 11).

3.4.1.2 Examinations at ANO and Oconee

Fork radiation detectors were used at Arkansas Nuclear One (ANO) and the Oconee Nuclear Station to verify utility reactor records.[12] The fork detectors provided a nondestructive means for characterizing the fuel assemblies following discharge and cooling time in the spent fuel pools. The fork detector analysis was conducted underwater in the spent fuel pool using a measurement taken at about the assembly centerline. [At ANO Unit-1 (ANO-1), a steel band located at the center of each assembly precluded a measurement at that level, so the measurement was taken 1 ft above the centerline of each assembly. Other measurements showed the neutron and gamma-ray yield to be essentially constant along the central section of similar PWR assemblies, extending a few feet on either side of the center.] The count time on each assembly was 100 s. The fork detector system used two arms that were placed on either side of the fuel assembly facing one another. Each arm of the fork detector contained two fission chambers for neutron detection and an ionization chamber for gamma-ray detection to provide information on the neutron and gamma radiation fields emanating from the SNF assembly being examined.[13] Because the gamma-ray detector is less sensitive to variations in burnup, it confirms burnup with approximately 15% uncertainty[9] and is thus used only as a general confirmation of assembly cooling time and burnup.

3.4.1.3 Measurement Data Analysis

The approach used in the analysis of the SNF at both sites was to accumulate measurements from a number of assemblies and generate an internal calibration by comparing each assembly with the best-derived fit of all the site data. The self-calibration (best-fit curve) eliminated the uncertainties and complications that were introduced by external calibration techniques while retaining the sensitivity to detect measurements that were inconsistent with the utility reactor records. The analysis of the fork detector data made use of the reactor records for cooling time, burnup, and initial enrichment in such a way that errors in any of these parameters were likely to increase the deviation from the best-fit calibration curve. During out-of-core examinations, any observed deviations from the best-fit curves would incorporate the uncertainties in the out-or-core measurements and any errors in the reactor records. The average deviations were therefore likely to be upper bounds on the random errors in the reactor records for assembly burnup.[9] To correct the observed data for the variation in cooling times among the assemblies, the neutron data (after background subtraction) were extrapolated back to the date of discharge of each assembly using an exponential factor of half-life equal to 18.1 years, which is the half-life of the principal neutron emitter, ^{244}Cm. A factor to adjust the observed count rates for the variation in initial enrichment among the assemblies was calculated using reactor records.[9]

At ANO-1, 34 assemblies were measured with the fork system in 1.5 days of operation. The initial enrichment of the assemblies ranged from 2.016 to 3.209 wt % ^{235}U. The range of assembly average burnup was from 19.9 to 57.3 GWd/MTU, and the cooling times varied from 6.1 to 17.6 years. Background counts were generally less than 1% of the signal from the assembly.[9,14] At ANO Unit 2 (ANO-2), measurements were made on 39 standard assemblies with average enrichments ranging from 1.9 to 3.9 wt % ^{235}U. The range in fuel assembly average burnup was 12.3 to 50.7 GWd/MTU, and the cooling times ranged from 3.8 to 13.7 years.[9] Two assemblies that contained neutron sources and had been cooled for approximately 2 years were also examined during this period. The fork detector system identified the presence of neutron sources in these two assemblies by a rise of 25 to 40% in the signal in the neutron detectors at the location of the neutron sources near the midpoint of the assemblies.[9]

The neutron and gamma-ray emissions in the examined assemblies were measured in the spent fuel pool by raising each assembly partially out of the storage rack and performing the measurement near the center of the assembly (see Fig. 3.2).

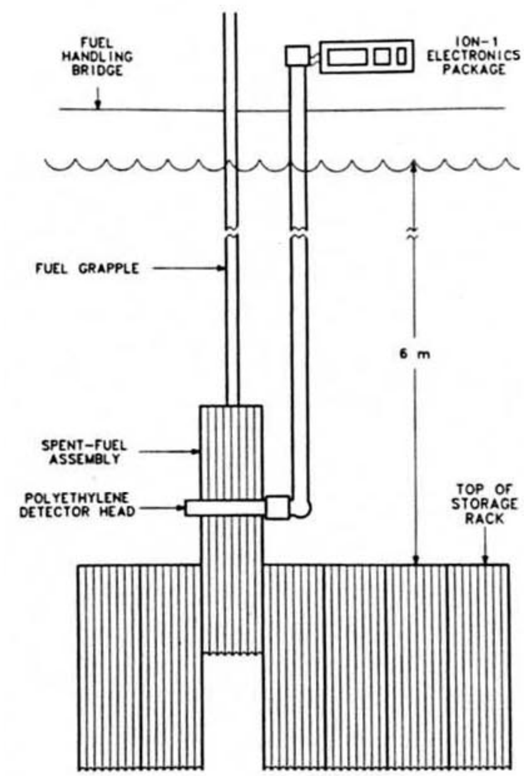

Fig. 3.2. Schematic showing the method for making measurements in a spent fuel pool using the fork detector system. (*Source:* Reference 9.)

The overall uncertainty introduced by the out-of-core measurements with corrections was about 2% of the reactor record indicated burnup. At ANO-1, the self-calibration curve was the power law best-fit (least squares) to the data and is given by

$$N = (C)\,(B)^{3.83} \,, \qquad (2)$$

where N is the neutron count rate in counts per second, B is the burnup in units of GWd/MTU, and C is a fitted constant whose value is 0.00100. At ANO-1, the neutron signal was proportional to the 3.83 power of the burnup, and at ANO-2, the neutron signal was proportional to the 4.35 power of the burnup. The average deviation of the burnup measurements from the best-fit curve of reactor record burnup vs neutron signal was 2.7% at ANO-1, with a maximum deviation for a single assembly of 9.1% (see Fig. 3.3). At ANO-2, the average deviation of the burnup measurements from the best-fit curve of burnup vs neutron rate was 3.5%, with a maximum deviation for a single assembly of 8.6%. This was consistent with the 2 to 3% random variation among the reactor records for average assembly burnup.[9]

Because the fork detector measurements were taken only at the assembly centerline and compared with a best-fit calibration curve, the true assembly full-length average burnup values were not actually measured. The average burnup values for the assemblies were developed as calculated numbers based on the fork detector response (measured counts) at each assembly centerline, comparison of the detector output with the calibration standard, and adjustment of the result for other factors such as assembly cooling times.

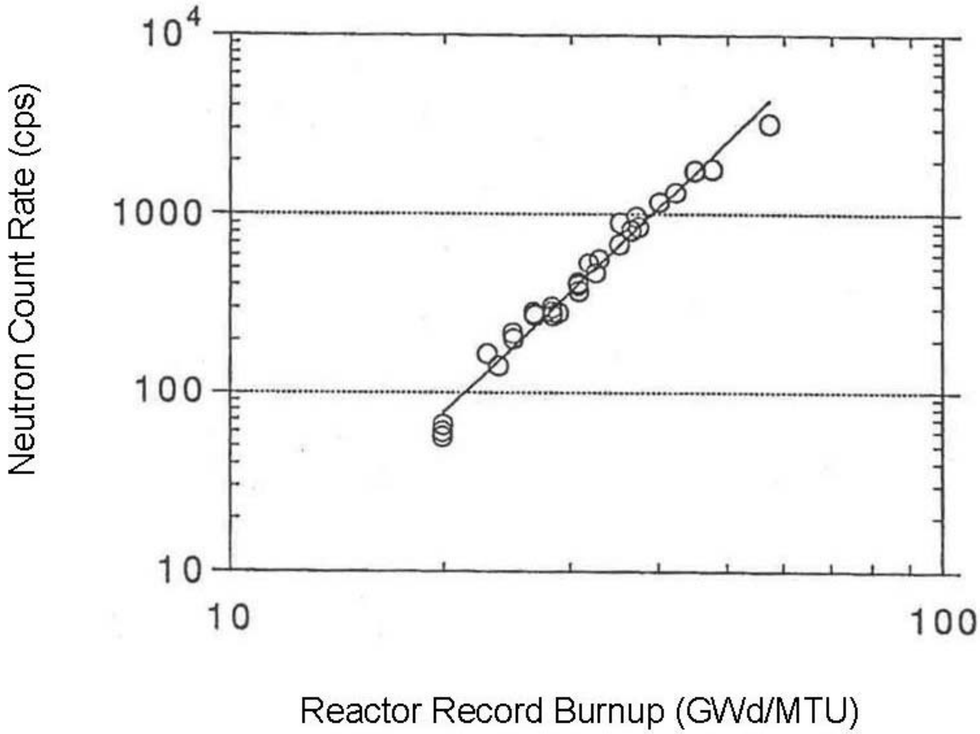

Fig. 3.3. Log-Log best-fit plot of neutron signal vs reactor record burnup at Arkansas Nuclear One Unit-1. (*Source:* Reference 9.)

Another fork detector demonstration program was conducted in 1993 at the Oconee Nuclear Station. In conjunction with Sandia National Laboratories and Los Alamos National Laboratory, Oconee staff used the fork detector system to measure the burnup of 93 assemblies from Oconee Units 1 and 2 over a 3.5-day period. The initial enrichment of the assemblies ranged from 2.91 to 3.92 wt % ^{235}U, and the average burnup ranged from 20.3 to 58.3 GWd/MTU. The cooling times ranged from 4.2 to 14.8 years. The measurements showed an average deviation in burnup from the best-fit curve of about 2.2% with enrichment correction and 10% without the correction. Among the 91 assemblies fit by the curve, only one assembly deviated by more than 6%. Two assemblies exhibited much higher (factor of 5) neutron signals than expected from the burnup records, and a check of the reactor records for these two assemblies found that they contained small americium-curium-beryllium primary neutron sources.[9]

During the fork measurement programs at ANO and Oconee, both Babcock and Wilcox (B&W) and Combustion Engineering SNF assemblies were examined. A finding from these measurement activities was that the burnup dependence of the neutron signal was specific to each assembly design. For the B&W assemblies at ANO-1 and the Oconee Nuclear Station, the neutron signal was proportional to the 3.83 power of the burnup. For the Combustion Engineering assemblies at ANO-2, the neutron signal was proportional to the 4.35 power of the burnup. Thus, a direct comparison of data between assembly types is possible, but only after assessing and correcting the data for the impact of the different assembly designs. This finding further demonstrated how these out-of-core examinations were dependent on accurate and detailed calibration and data interpretation.[9]

The Westinghouse measurement system used at the Surry Power Station in the early 1980s was similar to the fork detector system described above in that it used two fission chambers on opposite faces at the midplane of a fuel assembly and the associated signal conditioning, recording, and data analysis hardware and software. The measurement of an assembly in the reactor spent fuel pool took about 10 min from grappling of the assembly in the fuel storage rack; raising, moving, and placing it in the mounting fixture on the pool bottom; counting the assembly; and returning the assembly to its original location in the storage rack. Measurements were taken only at the midpoint of the fuel assembly. A feature of this measurement methodology is that the measurement at the midpoint of the fuel assembly would typically have the highest burnup instead of the assembly average burnup. When this work was conducted in the 1980s, it was assumed that no errors were introduced by this measurement procedure if the peak-to-average burnup ratio was a constant, independent of burnup. It was assumed that peak-to-average burnup ratios were not much greater than unity for high-burnup PWR fuels. Thus, the measurement of burnup in the midplane of the assembly would provide a good measure of assembly average burnup in high-burnup PWR fuels with relatively uniform axial power distributions.[4] However, near the fuel assembly ends, burnup is suppressed due to leakage. Consequently, the majority of PWR SNF assemblies have similar axial-burnup profiles (or shapes)—relatively flat in the axial midsection (with peak burnup of ~1.1 times the assembly average burnup) and significantly underburned fuel at the ends (with burnup of ~0.5 times the assembly average burnup).[15] Also, the burnup is slightly higher at the bottom of the assembly than at the top due to the difference in the moderator density. The cooler (higher-density) water at the assembly inlet results in higher reactivity (which subsequently results in higher burnup) than the warmer moderator at the assembly outlet. Other factors, such as control rods or axial power-shaping rods, can also affect the axial burnup shape. Thus, measurements at the midplane of a fuel assembly may be used to give a reasonable estimate of the assembly average burnup, but may also introduce uncertainty into the burnup calculation when determining assembly average burnup.[16]

In the demonstration at Surry, 50 fuel assemblies were measured, covering a burnup range of 14.41 to 41.05 GWd/MTU, a cooling time of 0.18 to 8.86 years, an initial enrichment range of 1.86 to 3.40 wt % ^{235}U, and a considerable variation in fuel power history and cross-assembly burnup gradients. The data were analyzed to provide correlations and correction terms for burnup, enrichment, and time dependence of the neutron source. The final analysis of results provided an estimated deviation of 0.8 GWd/MTU from utility-reported average burnup values on each assembly. Because the utility reactor records have an uncertainty of approximately 0.6 GWd/MTU, the neutron-counting technique appeared to provide an accurate and sensitive measure of burnup when enrichments and cooling times were known and the system was calibrated to identical assemblies of known characteristics.[4]

The Surry assessment of the neutron-counting system, which included evaluations of two detectors on opposite faces vs a single detector, found that the standard deviation nearly doubled with the single detector because of nonuniformity of burnup across assemblies and small positioning errors. The reproducibility of repeated count-rate measurements of the same assembly was also checked. A 6% variation was found for the shortest-cooled fuel, with a high gamma-ray background (0.18 years), but the variation appeared to be under 4% for a representative population of fuel.[4]

Using a reference burnup uncertainty of 2%, a count rate uncertainty of 6% (consistent with the 6% variation noted above), and a burnup coefficient of 4 (as determined in the Surry measurements and confirmed by ORIGEN2[17] data), the following formula was used as a basis to determine the burnup uncertainty at Surry:

$$B = B_r \left(\frac{C}{C_r}\right)^{\frac{1}{p}} \left(\frac{t}{t_r}\right)^{\frac{m}{p}} , \qquad (3)$$

where B is burnup, B_r is reference burnup, C is the net count rate, C_r is the reference net count rate, t is age, t_r is reference age, and p and m are analytically (or experimentally) determined coefficients for burnup and time dependence.

Solving the equation for uncertainty in burnup, the equation becomes

$$\left(\frac{\delta B}{B}\right)^2 = \left(\frac{\delta B_r}{B_r}\right) + \frac{1}{p^2}\left(\frac{\delta C^2}{C} + \frac{\delta C_r^2}{C_r}\right) + \left(\ln\left(\frac{B}{B_r}\right)\right)^2 \left(\frac{\delta p}{p}\right)^2 + \left(\ln\left(\frac{t}{t_r}\right)\right)^2 \left(\frac{\delta m}{p}\right)^2 . \tag{4}$$

The last two terms allow for the uncertainties in the sensitivity coefficients. However, because these terms also depend on the proximity of the burnup values and times of the measured and referenced fuel, they were assumed to be small (i.e., zero) relative to the other terms. The uncertainty in burnup was then calculated to be 2.9%. Based on the out-of-core measurement program, it was concluded that neutron counting of PWR assemblies in a system calibrated with identical assemblies of known burnup and age provided a very accurate estimate of assembly burnup. Another conclusion reached in these examinations was that the equipment sensitivity imposed significant requirements on the measurement process, because a lack of attention to measurement details could quickly generate very inaccurate results.[4]

3.4.2 High-Resolution Gamma-Ray Spectroscopy

In addition to the fork detector measurements noted above, ANO also performed five measurement campaigns using high-resolution gamma-ray spectroscopy to provide confirmation of the operator-declared burnup values. Pajarito Scientific Corporation (currently BNFL Instruments, Inc.) performed this work between April 1996 and August 1997. The first campaign was a demonstration campaign in which 52 assemblies were measured to test and refine the monitoring equipment. Four additional campaigns measuring 351 assemblies were later performed. The assembly types chosen were B&W 15 × 15 assemblies and Combustion Engineering 16 × 16 assemblies. These assemblies had a wide range of irradiation histories comprising various burnup values (12.31 to 46.87 GWd/MTU), cooling times (2.51 to 20.57 years), and enrichments (1.921 to 3.902 wt % ^{235}U).

The fuel measurement procedure for these HRGS examinations initially included instrument standardization by means of measuring a reference source. However, this produced large errors and exposed the test operators to an unnecessary amount of radiation. Instrument standardization was eventually achieved by measuring a selected reference assembly. The data collected from the five measurement campaigns were not presented in a published report. However, the rough draft analysis of the campaign 3 measurement data, as a sample set, suggested that the dependent calibration 1σ measurement uncertainty may be apportioned as 2.33% to the measurement process and 4.8% to the reactor record error to give the observed overall uncertainty, for campaign 3, of 5.37% at the 67% confidence interval. The distribution of the data was symmetrical and continuous. The determination of cooling time from the activity ratio ^{134}Cs/^{154}Eu was effective in its role in the burnup measurement procedure as a cooling time correction parameter. The average cooling time uncertainty was less than 120 days. The uncertainty was calculated from a combination of a dependent calibration based on the correlation between operator-declared data and assembly-measured cooling times and statistical uncertainties.

During the HRGS campaigns, the standard measurement time for the gamma-spectra acquisition was about 15 min per assembly. The total measurement time, including fuel-handling time, was approximately 1 h per assembly. The handling of the selected fuel assemblies for testing was an off-normal event. To minimize the background source strength, the fuel assemblies had to be moved to a location in the pool that was shielded from the bulk of the fuel assemblies. At ANO, the testing was performed in the cask pit. Fuel assemblies had to be moved to the point where the fuel rod would touch the detector. The detector was used as a proximity indicator. In this configuration, there were no limit switches that could be used to limit the movement of the fuel-handling machine to prevent inadvertent damage to the fuel assemblies. The examination relied heavily upon operator actions. During the examinations, the fuel assemblies had to be moved downward along the detector to allow acquisition of gamma-ray spectra from a number of points along the length of the assembly. The assembly was subsequently rotated 180° about its vertical axis, and a repetition of the gamma-ray scanning was conducted. The combination of the scanning and rotation of the assembly maximized the sampling of the assembly and minimized the accumulation of systematic errors arising from axial or radial burnup variations due to the axial burnup profile or radial tilt. Assembly hang-ups did occur during these activities. Additionally, rotating the fuel assemblies was difficult and was not a normal fuel-handling practice. During the fuel moves, the operating team received radiation doses even when examining spent fuel assemblies that had long cooling times and relatively low burnup values. Based on the potential for fuel-handling problems, ANO performed a risk, dose, and cost assessment to evaluate the measurement test program. The evaluation suggested that due to the high risk of fuel damage and additional dose and the high cost of the examination program, the value associated with performing additional measurements was not sufficient to justify continuing the measurement program. At that point, the measurement campaigns at ANO were halted.

However, before the gamma-scan program was halted, three of the ANO-1 fuel assemblies were identified as "outliers" by the detector system. Subsequent evaluation of the reactor records for those assemblies by ANO staff revealed the possibility that the examination responses may have been perturbed by mistakes in the data correlation. Thus, the measurements for these assemblies were considered to be flawed, and the reactor burnup records for the three assemblies were assumed to be correct.

In another documented examination program, Zion Station Reactor 2, cycle 1, data were benchmarked for the Electric Power Research Institute (EPRI) Advanced Recycle Methodology Program (ARMP). Assemblies with enrichments of 2.25% (Region 1), 2.79% (Region 2), and 3.29% (Region 3) ^{235}U and burnup values averaging 19.95, 19.17, and 14.43 GWd/MTU, respectively, were measured using HRGS based on ^{140}La decay. Direct burnup accuracies were not calculated for the assemblies at the time the data were collected but were later inferred from the data. A subsequent evaluation showed an uncertainty of less than 2% between the utility-calculated burnup and the gamma-ray spectroscopy-measured burnup.[4,18]

3.4.3 Summary of U.S. Out-of-Core Measurement Results

None of the systems used for out-of-core measurement of SNF burnup actually measure assembly fissile content directly. Instead, such systems measure gamma-ray or neutron emissions from nonfissile materials in an assembly and then use that information to calculate the assembly fissile content. There have been a number of measurement campaigns using various types of equipment, with the majority of the examinations using the fork detector system. The available data comparing the use of either HRGS equipment or neutron-counting fork detector equipment with utility reactor records are significant but not overwhelming. The neutron-counting systems appear to have more accuracy and are less limiting than the gamma-ray scanning systems, but they still require the use of reactor record data for burnup, cooling time, and initial enrichment as input parameters. Also, the fork detector system requires a geometrically identical reference assembly of known burnup and cooling time or the development of a best-fit curve for calibration purposes before any examinations are conducted. The "known" burnup of the reference assemblies is taken directly from utility reactor records.

Several of the referenced reports state that the deviation of fork detector system measurements from utility records is in the 2.2–5% range.[4,9] The specific results noted in Section 3.4.1 of this report show the average accuracy of the fork detector when compared with reactor records was between 3.0 and 3.5%, with maximum assembly deviations of up to 8.6%. Even though the gamma-ray scans were conducted on the full length of the fuel assemblies instead of the fuel centerline as was done in the fork detector examinations, the high-resolution gamma-ray scans were not as accurate as the fork detector examinations. Furthermore, in ANO campaign 3, the gamma-ray scans demonstrated an average deviation from plant records of 5.37% at the 67% confidence interval.

The accuracy of out-of-core measurement systems used for demonstration measurements is adequate for the purposes of verifying that fissile content has not been removed/diverted from an assembly (the purpose for which the equipment was developed) or for ensuring that a significantly underburned fuel assembly is not mistaken for a fully burned assembly and subsequently placed in a transport cask. Review of the various measurement campaigns indicates that these out-of-core measurement systems are somewhat less accurate than reactor records and, as noted, the results of the out-of-core examinations are highly dependent on reactor records. For any out-of-core measurement campaign, the as low as reasonably achievable (ALARA) radiation dose implications for the examination personnel, the value added by the measurement campaign, the potential increase in fuel mishaps, and the potential impacts on other spent fuel pool operations should all be considered.

Since the out-of-core burnup measurement programs were conducted at Surry and ANO-1, the equipment vendors have upgraded their equipment to incorporate lessons learned and improve the sensitivity of the equipment; however, the upgraded equipment has not been demonstrated at any U.S. commercial utility to measure SNF assembly burnup. A description of the upgraded systems is provided in Table 3.2.[19] Another different measurement approach that is potentially relevant to burnup confirmation, albeit approximate, is the measurement of light intensity from Cerenkov radiation, which is discussed in the following section.

Table 3.2. Burnup verification systems comparison

	Westinghouse	BNFL	EPRI/Sandia
System Name	None specific	None specific	FORK+
System Description	A miniature SiC nuclear detector designed for use in harsh radiation and temperature environments. SiC has several advantages for use as a solid-state radiation detector material, primarily due to its wider band-gap energy compared with that for silicon or germanium: 1. Lower leakage current 2. Operational capability at elevated temperatures 3. Greater radiation resistance 4. Capable of pulse-mode operation in high-radiation fields Gamma rays and neutrons can be measured simultaneously. Demonstrated linear response to both neutron and gamma-ray fluxes. Detector sensitive to thermal or epicadmium neutron energies. Does not require external cooling.	An HRGS system that uses a high-purity Ge detector. The detector requires nitrogen cooling to operate. The system demonstrated in the United States measures only gamma radiation. However, neutron techniques (fission chambers) have been incorporated into the system used at Sellafield (Thorp). Demonstrated linear response to gamma-ray fluxes.	A gamma-ray spectroscopy system which uses a CdZnTe (CZT) detector. CZT crystal requires tungsten shielding to improve resolution. In addition, a cadmium liner is required over the tungsten to absorb thermal neutrons to keep the thermal neutron fluence at the crystal to a minimum. (Note: Neutron absorption by Cd produces a gamma ray; therefore, the CZT crystal must be protected.) Neutron measurement is accomplished through the use of a fission chamber. The CZT crystal is cooled thermionically by using a microelectronic thermionic cooling device. Larger CZT crystals would require nitrogen cooling.
Detector Types	SiC (^{6}LiF coated)	High-purity Ge	CdZnTe with tungsten shielding and lined with cadmium.
Source of Signal	Gamma rays Tritons—Neutron detection is achieved through the juxtaposition of an enriched ^{6}LiF layer. Triton particle detection is used to infer neutron flux. Alpha particles may be stopped by an Al absorber to minimize SiC radiation damage.	Gamma rays	Gamma rays Neutrons

Abbreviations: BNFL = British Nuclear Fuels plc; EPRI = Electric Power Research Institute; HRGS = high-resolution gamma-ray spectrometry; Thorp = Thermal Oxide Reprocessing Plant. (*Source*: Modified from Appendix A in Ref. 19.)

3.5 CERENKOV RADIATION MONITORING DEVICES

Cerenkov radiation is emitted whenever a charged particle passes through a medium at a velocity exceeding the phase velocity of light in that medium. In water, the phase velocity of light is about 75% of its value in a vacuum. An electron passing through water and having a kinetic energy greater than approximately 0.26 MeV will produce Cerenkov radiation. In spent fuel, these electrons include Compton electrons produced by gamma radiation, beta rays that escape directly into the water, and the interactions of high-energy neutron capture gamma rays that produce electrons from Compton scattering and pair production. The intensity of Cerenkov light generated by irradiated fuel is proportional to the radiation field intensity in the vicinity of the irradiated fuel. This field intensity is proportional to the burnup of the fuel and inversely proportional to the fuel cooling time.

Cerenkov light measurement is a very simple and relatively nonintrusive method for verifying the effectiveness of spent fuel safeguards compared to other methods because it involves only the viewing of spent fuel assemblies using a Cerenkov viewing device (CVD), and no movement of the stored spent fuel is required. Current CVDs can detect Cerenkov glow images without interference from normal lighting. Because Cerenkov emissions range from the ultraviolet (UV) to the infrared, the CVD uses a UV light image intensifier, which has good sensitivity to UV light due to the use of a Cs-Te photocathode material. The CVD also uses a conventional night viewing device with an optical filter to minimize interference from lights in the facility. Use of modern CVD equipment allows the detection of a weak Cerenkov glow image that may result from fuel that has a low burnup or a relatively long cooling time.[20]

An improved CVD, the UV-I.I CVD, was tested in Japan at BWR and PWR facilities in the early 1990s with fuel burnup values ranging from 6.2 to 33 GWd/MTU and cooling times from 370 days (1 year) to 6,300 days (17.2 years). The CVD was placed in a waterproof housing with a window and was submerged in the spent fuel pools to about 10 cm. The distance between the CVD and the spent fuel was about 7 m, and all facility lights were in their normal (on) position.[20] The Cerenkov radiation occurs strongly in the space between fuel pins and is highly collimated when viewed along the axis of the assembly. This is the normal view for LWR fuel because the spent fuel is stored vertically and is viewed from above the spent fuel pool. The highly collimated nature of the light requires that the instrument used to measure the Cerenkov glow have its optical axis positioned along the axis of the fuel assembly to view the maximum light intensity emitted by the assembly.[21] The results of the tests in Japan showed that the improved CVD was capable of providing clearer and more vivid images of the spent fuel than older CVDs and was able to identify Cerenkov patterns in the spent fuel for low-burnup and long-cooling-time assemblies.

In 1993, the Swedish and Canadian safeguards support programs began a joint program to develop a high-sensitivity Cerenkov viewing device. This instrument was called a digital CVD (DCVD). In January 2002, a prototype DCVD was tested at Central Interim Storage for Spent Fuel in Sweden on PWR fuel and nonfuel, long-cooled BWR fuel, and Ågesta test reactor fuel assemblies.[22] Six PWR fuel assemblies were measured to determine their Cerenkov intensity as a function of cooling time. The burnup values and cooling times of the six fuel assemblies are shown in Table 3.3. Intensities were measured by selecting the brightest pixels (less the brightest 1% of intensities to reduce the effects of noise). Since theoretical calculations of photon intensities as a function of burnup and cooling times for PWR fuel are not available, corrections to normalize the higher and lower burnup values (T12 and K30, respectively) could not be done. The authors concluded that the prototype DCVD was successful in meeting their measurement objective of verification of fuel with a burnup of at least 10 GWd/MTU and a cooling time of 40 years or less.[22]

On the basis of the six PWR examinations, the authors of the Swedish report indicated that it may be possible to determine the cooling time of an assembly knowing its burnup (from reactor records) and DCVD Cerenkov intensity. Considerable work remains to be done in this area, specifically with respect to the precision of readings and the stability of the CVD detector. Water quality in spent fuel pools is also an important factor, although it should be constant in any one facility over a short time period.[22] It was noted that the spent fuel pool at Ringhals Unit 2 contained about 2,000 μg/g boron to absorb neutrons emitted from the spent fuel. The boron caused the light intensity of the Cerenkov glow to be reduced to about half the intensity received from similar spent fuel assemblies in a pool with no boron.[22]

Table 3.3. Intensity measurement of six fuel assemblies

Fuel identification number	Burnup (GWd/MTU)	Cooling time (years)	Intensity (counts)
Z04	44	1	17101
Y26	43	2	8113
W05	43	4	4053
T12	56	6	2891
R26	41	9	1047
K30	32	14	462

Source: adapted from Reference 22.

Another consideration that must be taken into account is the "near-neighbor" effect of spent fuel assemblies in close proximity to the assembly being examined (either adjacent to or diagonal with the assembly being examined). The near-neighbor effect occurs when gamma rays from neighboring fuel assemblies travel into the adjacent assembly and generate Cerenkov light in the water spaces. In BWR Cerenkov examinations, a clear near-neighbor effect has been noted; in the limited PWR examinations conducted in Ref. 22, the data were mixed; thus, a clear correlation showing how much the near-neighbor effect has on PWR spent fuel assemblies is still unknown.[22]

As of 2007, the DCVD is in its third generation, has integrated hardware, is equipped with a zoom lens (80 mm–200 mm), and uses windows-based software. During a recent training course at IAEA, the new DCVD was tested, and the DCVD partial defect test capability was successfully "demonstrated" to perform improved safeguards functions (detect single rod defects).[23] The authors noted that the DCVD could not clearly identify unirradiated substituted rods in spent fuel assemblies, indicating that the Cerenkov glow examinations would still not provide satisfactory information for lightly burned or very old spent fuel.

In summary, equipment and techniques exist that can be used successfully to detect the Cerenkov light given off by spent fuel; however, CVDs were developed to support nuclear safeguards programs and have not been optimized to measure spent fuel burnup except in a broad sense. For low-burnup fuel or fuel cooled for significant time periods, the Cerenkov glow will be very dim—although it should still be detectable. Equipment alignment is critical to this type of examination and is especially critical when examining fuel that has been cooled for long periods of time. Near-neighbor assemblies and pool water quality (boron concentration) can also affect the examination results. It may be very difficult to distinguish between old, moderately burned fuel and new, very lightly burned fuel by the Cerenkov glow measurement. Nevertheless, a Cerenkov glow examination may be capable of providing "go/no-go" information for imprecise screening.

3.6 COMMENTS ON INTERNATIONAL PRACTICES AND REGULATIONS

The issue of how to demonstrate compliance with safety criteria, for example, minimum required burnup, when applying credit for fuel burnup is of common interest internationally. As evidence, during the most recent IAEA technical meeting[24] on burnup credit, a working group was convened to review the status of methods used to demonstrate compliance with safety criteria among nations currently applying burnup credit. The main observations of the working group report[24] are summarized in this section, which also includes recent information on German regulations and practice; the reader is referred to the working group report for the full details.

The working group discussions focused on methods of verifying assembly burnup and reviewed examples of how this is put into practice as part of spent fuel operations in France, Germany, the United Kingdom, and the United States. The examples highlighted both similarities and differences in practice. Where differences were identified, the group reviewed the causes both in terms of variations in operational design/requirements and in terms of any differences in the underlying safety philosophy. In general it was found that the safety philosophy and associated methods are very similar in all the countries reviewed and that differences with respect to compliance issues arise mainly from differences in the operational environments. Examples of spent fuel operations reviewed by the working group are provided below.

An outcome of the working group discussions was a recommendation that the IAEA Standard on Transport[25] be reviewed with respect to the current "requirement" for a measurement of burnup. The recommendation stems from the recognition that the rigor required to demonstrate compliance of any safety criterion should be dependent upon the importance of that criterion to the overall safety, which can depend on many aspects of the spent fuel system and its operations. A related observation is that there is significant variation between standards with respect to whether a measurement of burnup is a firm requirement or not.[*]

3.6.1 Summary of Current Practices and Regulatory Requirements

The following examples of burnup credit practice were presented and discussed by the group.

- Dissolution in the Thorp reprocessing plant (burnup credit is not used for on-site storage)
- Transport, storage, and dissolution at the Cap de La Hague reprocessing plant
- Reactor pool storage in the United States and Germany

3.6.1.1 United Kingdom—Thorp Reprocessing Plant

Criticality evaluations for transport of SNF to the Thorp plant are based on the fresh-fuel assumption. Burnup credit is applied to the head-end plant, particularly in the dissolvers, where large batches of fuel comprising several fuel assemblies are sheared into small lengths and dropped into hot nitric acid doped with gadolinium. The original criticality assessment was based on the fresh-fuel assumption, which resulted in a requirement for significant concentrations of gadolinium to be added for criticality control. The current criticality assessment is based on actinide-only burnup credit, which reduces the gadolinium addition requirements (and hence reduces waste volumes).

Consistent with typical practice, the safety criteria are based on a 5% administrative margin for normal conditions with additional allowance for code bias and uncertainty (including uncertainty associated with the burnup credit approach). For some low-probability-accident conditions, a reduced administrative margin (2% in k_{eff}) is applied. The burnup credit evaluation includes numerous conservative assumptions; for example, the fuel packing fraction in the dissolver is optimized to maximize the calculated k_{eff} value, which is reported to represent an additional margin of about 14% in k_{eff} relative to typical packing fractions.

Compliance with the loading curve is made through a combination of checks against the supplier's data and through measurements made on each assembly by the Thorp fuel pond feed monitors. The measurement is based on gamma-ray spectroscopy and neutron counting, which provide information on

[*]It was noted that while the IAEA standard requires that "a measurement shall be performed," (paragraph 674 of [25]), the advisory material explains that the "measurement technique should depend on the likelihood of misloading the fuel and the amount of available subcritical margin due to irradiation."

cooling time, burnup, initial enrichment, and residual enrichment (RE).*[26] A go/no-go trip is set against the measured RE value, which prevents any assembly above the limiting RE from being fed forward to the shearing machine. The assessment demonstrates that the RE at the zero burnup end of the loading curve is bounding (i.e., a minimum) for the rest of the curve, so this value, with allowance for measurement and calibration uncertainties, is used as a test of compliance for all assemblies. The calibration is made through measurements on selected fuel assemblies before each campaign, and the measurements are calibrated against supplier data. Checks are made during and after each campaign to confirm calibration constants and uncertainties. The measured RE is based on neutron counts taken at a single axial location near the center of the element.

Based on the available information, it is surmised that this system is more of an approximate check of burnup, through the determination of the RE, than it is an accurate verification or quantification of burnup.

3.6.1.2 France—Transport, Storage, and Dissolution at the Cap de La Hague Reprocessing Plant

At the Cap de La Hague reprocessing plant, burnup credit is applied for on-site pool storage of PWR SNF and dissolution in the rotary dissolvers, where sheared rods are dropped into one of the dissolver buckets that are soaking in hot, unpoisoned nitric acid. For these two applications, in which the required level of credited burnup is different, the burnup credit approach is also noticeably different. For transport of SNF to La Hague, the criticality assessments are also based on burnup credit, very similar to those applied for pool storage. For all applications, the safety criteria are based on an administrative margin of 5% in k_{eff} for normal conditions and some accident conditions, with additional allowance for code bias and uncertainty. For some unlikely accident conditions, the administrative margin is reduced to 2 or 3% in k_{eff}. Some uncertainties associated with burnup credit are taken into account through additional conservatisms in the depletion calculations.

The original assessments for storage and transport were based on the fresh-fuel assumption, which was sufficient to comply with the criticality safety criteria for low enriched fuels. However, as fuel enrichments have increased, the assessments have become based on actinide-only burnup credit and the following two conditions relative to burnup verification.

- If the minimum required burnup is less than or equal to 3.2 GWd/MTU, a simple gross gamma-ray measurement (or a validated—by regulators—equivalent method to confirm the irradiation) is sufficient.
- If the minimum required burnup is higher than 3.2 GWd/MTU, a validated (by regulators) burnup measurement must be performed. The intention of this measurement is to verify that the irradiation in the 50 least-irradiated centimeters (axially) of the fuel assembly is higher than the minimum required burnup.

These burnup verifications are performed before loading the fuel assemblies into a transport cask and are completed by the supplier. The type of measurement, made in the supplier's plant, depends on the supplier (French, German, etc.) but must be approved by the French regulators.

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*RE is derived from measurement of sub-critical neutron multiplication, calibrated against data received from the suppliers of the fuel. These data are used to form the equivalent ^{235}U enrichment from the residual ^{235}U, ^{239}Pu, and ^{241}Pu in the fuel. For ^{239}Pu and ^{241}Pu factors are applied to create their equivalent (in reactivity) ^{235}U mass. These factors are based on the relative thermal fission cross-sections. The values of RE derived from the supplier data is then used to convert the measurements of neutron multiplication into RE.

For dissolution in rotary dissolvers, the safety assessment is based on actinide-only burnup credit, which leads to loading curves: maximum permissible mass per dissolver bucket as a function of burnup (one loading curve for each initial enrichment). Each of these curves presents a burnup limit over which criticality safety is ensured by the bucket geometry (without restrictions on loaded mass). A PWR fuel assembly is typically loaded into three or four buckets (about 110-liter volume each).

Compliance with the loading curves and determination of the number of buckets needed to load a fuel assembly are made through a combination of checks against the supplier's data and through measurements made on each assembly. The measurements must provide information on the initial enrichment, the average burnup, and the assembly axial profile. These calibrated and validated measurements are implemented between the storage pools and the dissolvers and consist of gamma-ray scanning (on two opposite faces of the fuel assembly) and passive neutron measurements (on the two other opposite faces). The axial scannings are interpreted by an online evaluation program. A go/no-go trip is set against the comparison between the average burnup measured and the supplier's data. Details were not provided on the measurement calibration and associated uncertainties.

3.6.1.3 Germany—Transport and Storage

The requirements for pool storage facilities for LWR fuel assemblies are contained in the German safety standards KTA 3602 and DIN 25471.[27] Transport and storage cask requirements are defined in the German safety standard DIN 25712.[28] A review of these German regulatory standards is available in Ref. 29. Although they are licensed under different regulations, the requirements in these regulations with respect to burnup credit applications to pool storage and transport/storage casks are completely consistent with respect to implementation and validation of the calculations, determination of criticality safety acceptance criteria and loading criteria, and prevention of misloading events. An administrative margin of 5% is applied for normal conditions, with allowance for a lower value under certain conditions.

Both DIN 25471 (Ref. 27) and DIN 25712 (Ref. 28) require that the misloading event has to be excluded as a design basis event by applying the double contingency principle directly to the misloading event: at least two independent, unlikely, and concurrent incidents have to happen before a misloading event can occur. This application of the double contingency principle and hence the exclusion of the misloading event as a design basis event from the criticality safety analysis is achieved by applying independent layers of hardware and software measures, ensuring the reliability of the reactor record data and the fuel-handling procedures applied to the pool and cask loading operations.[30] Therefore, no burnup measurement is required.

For transport/storage casks, DIN 25712 (Ref. 28) allows application of actinide-plus-fission-product burnup credit to LWR uranium oxide and mixed oxide fuel. Flooding of the transport casks, as well as of the storage casks, must be considered. In contrast to pool storage, a burnup verification measurement for fuel to be loaded in a cask is stipulated, per paragraph 674 of IAEA TS-R-1.

In the past, when the spent fuel was transported to France for reprocessing, a burnup measurement was performed according to the French requirements: measurement of the shape and burnup of the least-burnt 50 cm at the top end of the fuel zone for each assembly. For this purpose the Python system was used at the Brokdorf (PWR) and Gundremmingen (BWR) plants. Spent fuel shipments to France no longer occur. Under current German regulations, only a check (e.g., by performing a gross gamma-ray measurement) is required for SNF with record burnup values less than 10 GWd/MTU (i.e., less than one cycle burnup), which requires neither a calibration against reactor records nor an evaluation of uncertainties.

The following text, extracted from Ref. 29, provides particularly relevant insights and information relative to the German regulations and practices.

2.6.2. Quantification and verification of the burnup of the fuel assemblies to be loaded in a transport or dry storage casks (DIN 25712)
As laid down in the safety standard DIN 25712, quantification of the burnup of the fuel assemblies shall be based on the *evaluation of the reactor records*. Verification of the burnup shall be based on the *evaluation of the reactor records* as well and, additionally, on a *consistency check by means of a measurement* (such as a gamma scanning, a measurement of passive neutron emission, or a combination of both measurement procedures). Determination of the fuel's burnup and its verification shall be performed in compliance with the quality assurance requirements laid down in the German safety code KTA 1401 (cf. section 2.6.1).

The only reason that the requirement for a consistency check by means of a measurement has been included in the safety standard DIN 25712 is that this requirement is laid down in the IAEA regulation IAEA TS-R-1 [6]. If this were not the case this requirement would not have been included in the standard DIN 25712 since the evaluation of a measurement requires information from the reactor records [7] and since studies like [8] have shown that the utility-supplied data on burnup are of greater accuracy and reliability than could be provided by additional radiation measurement of spent fuel. The following rules are therefore laid down in the safety standard DIN 25712:

- The check whether the burnup of a fuel assembly fulfills the loading criterion (loading curve, cf. Fig. 6 for example) is carried out by using the reactor record information only: The loading criterion is met when the assembly's burnup obtained from the reactor records does not fall below an upper discrimination limit which is calculated, at a significance level of 5%, from the minimum required burnup given by the loading criterion and the uncertainty of the burnup value obtained from the reactor records. An example for determining such an upper discrimination limit is given in the informatory part of the standard DIN 25712.

- The consistency of a measurement result and the reactor record information is proven when the measurement result falls into an interval which is given by the lower and upper discrimination limit calculated, on the basis of a 5% significance level, from the reactor record information, the uncertainty of this information and the uncertainty of the measurement result. An example for determining such an interval is given in the informatory part of the standard DIN 25712.

- If it is obvious that the actual burnup of a fuel assembly is much greater than the minimum required burnup given by the loading criterion, then, instead of the consistency check, a measurement procedure may be used which is capable of demonstrating compliance of the assembly's burnup with the loading criterion. (Such a measurement procedure can be based for instance on the measurement of the intensity of the 662 keV gamma-transition following the decay of the burnup indicator Cs-137 [7][9].

3.6.2 Summary

Although approaches to burnup credit criticality assessment are very similar, the outcomes with respect to compliance procedures, particularly with regard to requirements for burnup verification measurements, vary considerably. In general the level of reliance placed on burnup verification depends on

- the amount of burn-up being credited;
- the level of confidence in other sources of information (e.g., reactor records);
- the presence of other contingencies (e.g., loss of control over boron concentration in a storage pool, flooding in a dry transport cask);
- the system and/or operational sensitivity to burnup and/or misloading; and

- the presence of other margins of safety not explicitly credited in the assessment (e.g., packing fraction in dissolvers, soluble boron in spent fuel pools).

A common comment during the working group discussions was that the requirement for a burnup measurement in their regulations and/or practices is necessary for compliance with the requirements of IAEA TS-R-1.

4 REACTOR BURNUP RECORDS: GENERATION AND ACCURACY

4.1 BACKGROUND

Reactor operation records are generated, controlled, and archived per plant procedures that support 10CFR50 Appendix B and are required to be readily retrievable for design, audit, or investigation activities. These records, also called core-follow data, contain operating parameters that are important to the safe and reliable operation of the nuclear plant and are either measured, calculated, or inferred from a combination of measurements and predictions.

Measured records include reactor operation monitoring data such as reactor power level, reactor coolant flow, and reactor coolant boron concentration. These data are obtained from plant instruments—thermocouples, flowmeters, radiation detectors, etc.—and electronic equipment that are calibrated and maintained per plant procedure to ensure a high degree of reliability and accuracy. Measurement uncertainties associated with these data have been quantified from extensive fuel vendor and utilities studies, which were approved by NRC as part of the reactor design and safety analyses methodology review.

Most core physics parameters, including those used for Technical Specifications compliance, cannot be directly measured and are either calculated by a nodal simulator neutronics code or inferred from testing involving reactor manipulation before cycle startup. These include control rod reactivity worth, soluble boron worth, moderator temperature coefficient, and shutdown margin (SDM). Three dimensional core power distribution can be inferred directly from neutron flux measurements, and depending on the reactor type, this measurement is online (continuous) or offline (flux mapping performed monthly or as needed). Typically BWRs, and some PWRs, have fixed in-core detectors, which allow for online core power distribution and thermal limits monitoring. Most PWR plants, however, do not have fixed detection systems and perform monthly flux maps using movable detectors that are inserted in the core. In both fixed and movable detection system plants, the flux is measured in only one-third or so of the fuel assemblies in the core. A computer code, using data generated by the reactor reload design codes, is used to calculate the power distribution for the entire core using the instrumented location measurements to infer the predictions in the noninstrumented locations.

The inferred core power distribution obtained from the combination of in-core measurements and corrected predictions is referred to by the plant engineers as the "measured" core power distribution. It should be noted that this characterization would be closer to correct if all the core locations were instrumented. Actually, only about 30% of a PWR core is instrumented. The decision to not instrument the entire core was made during the design phase of the previous generation of reactors and is based on the argument that instead of measuring all the fuel assemblies in the core, it is acceptable—using basic statistics—to directly measure the neutron flux in only a randomly selected representative sample of the core. Cycle-specific, burnup-dependent computer calculations are used to generate code bias and assembly-to-assembly coupling factors that are used to infer assembly average power and power peaking factors for uninstrumented assemblies based on the flux measurements in the nearby instrumented assemblies. This is accepted practice as the sample size is significant compared to the total fuel assembly population in the core. Also, due to the permanence and distribution of the core instrumented locations, a majority of the fuel assemblies will be measured for at least one cycle during their residence in the core as their locations are changed from one cycle to the next. A typical fuel assembly resides in the core for two to four reactor cycles.

Assuming the reactor core power distribution is adequately determined, it is then important to determine an accurate burnup value for each fuel assembly. The fuel assembly burnup is simply the amount of cumulative energy generated per weight of fuel. The burnup cumulated in fuel assembly i, ΔB_i, during a reactor operating time interval Δt can be expressed as

$$\Delta B_i = [(RTP * \Delta t * CF) / M_F] * P_i , \tag{5}$$

where ΔB_i is in megawatt–days per metric ton of uranium and Δt is in days. RTP is the reactor rated thermal power, a constant for a given core, expressed in megawatts, CF is the Capacity Factor, or the fraction of measured reactor thermal power to rated thermal power (CF=1 for a reactor operating at full power), M_F is the mass of fuel in metric tons of uranium, and P_i is the relative power in Fuel Assembly i.

In Eq. (5), M_F is precisely measured in the fuel fabrication facility and is formally transmitted to the nuclear utility. CF is determined accurately as the operating reactor thermal power is continuously measured at the plant by redundant methods. Because the measured thermal power is important to the determination of burnup, a discussion on its measurement accuracy is presented in Section 4.2.

All parameters in Eq. (5) are directly measured except P_i. Therefore, an accurate determination of the fuel assembly burnup requires accurate determination of the core power distribution.

The fuel assembly burnup is a core-follow parameter that is particularly visible to plant personnel and is currently used in regulation, procedure compliance, and design activities. The nuclear utilities are required by existing NRC regulation to keep the peak rod average burnup under 62,000 MWd/MTU for fuel integrity protection. Depending on the fuel assembly design, this rod-averaged limit corresponds to an average bundle peak burnup limit of about 45,000 MWd/MTU. The burnup records are also used for storage of discharged fuel assemblies in the spent fuel pool and in dry casks when they complete their core residency. The inferred burnup information is also an input to the special nuclear material (SNM) program. In current industry practice, the burnup values are used as input to the core design and safety analyses of the upcoming cycle as the starting exposure values for the reinserted assemblies, which typically compose two-thirds of the new core. The burnup values of these reinserted assemblies are confirmed from measurements performed during the power ascension tests following the refueling outage.

4.2 THERMAL POWER MEASUREMENT

As stated in Eq. (5), to calculate assembly burnup, the thermal power of the reactor must be known. The thermal output of a nuclear reactor is determined from temperature and flow measurements of the cooling water circulating through the reactor core. Thermal output is based on a complete steady-state energy-rate measurement across a defined envelope that includes the nuclear steam supply system. The primary components of thermal output are the energy rate in steady-state steam flow as measured by steady-state feedwater flow and the enthalpy difference between the steam (temperature and quality measurement) and the feedwater (temperature). Measured energy losses and credits are included for blowdown; letdown and makeup; seal water; component cooling water; electrical power inputs to pumps, heaters, and miscellaneous equipment; and radiative and convective losses. Standards from the American National Standards Institute (ANSI) and utility Technical Specifications detail the required accuracy of the flow measuring, temperature, and pressure devices in the nuclear steam supply system. The net result of these requirements is an overall core thermal output uncertainty of somewhat less than 1% at the time of measurement.[31] The effects of transients, below-full-power operations, and the possibility of instrument calibration drift between periodic calibrations all tend to increase the uncertainty associated with power-level measurements. Thus, the net accuracy of core power measurements is about 1% for the operational power-level measurements that are a primary input to fuel burnup determination.[4,18] The sum of the thermal power of all assemblies must equal the total thermal power of the reactor over any time span. Therefore, if the burnup of one assembly is high, another assembly burnup must be low, resulting in a "zero sum" characteristic of any errors. The most likely source of a possible error, one that applies to all assemblies, is the integral of the power of the reactor, which, as noted above, is measured very accurately.[4] The measurement of total core thermal power for calculation of burnup increments is done at least daily. The statistical error in total measured power from, for example, 100 days, gives a factor of 10 reduction in the single 1% uncertainty, with the result that the burnup uncertainty contribution from the measurement uncertainty associated with total core power is less than 0.1%.[32]

The time integral of the thermal power [in gigawatt-days] produced by the reactor is then used as the basis for the burnup assignment to individual assemblies using the SIMULATE-3,[33] or similar, computer code. The burnup assignment based on a core irradiation analysis that matches the actual energy output is often referred to as the "core-follow" assembly burnup.[34]

4.3 IN-CORE POWER DISTRIBUTION MEASUREMENT

Because of the differences in reactor types and designs, the in-core power measurement systems encountered in the nuclear industry are varied. However they share several common features, and their purpose and general layout are common to all units.

Fission chamber detectors are used to directly measure neutron reaction rates at several axial locations along the length of the fuel assemblies in the reactor. A typical fission chamber detector consists of two concentric cylinders, which act as electrodes. The collecting electrode is the inner cylinder and is separated from the outer electrode, which is coated with highly enriched U_3O_8 (90% ^{235}U), by a small gap. The gas in the gap is ionized by the charged particles resulting from neutron fission in the uranium. The negative ions are accelerated to the collector by a potential difference maintained between the two electrodes, and the resulting current is measured and is proportional to the neutron flux at the detector location. The number of detectors that are used varies and depends on the size of the reactor and whether the detectors are fixed inside the core (BWRs and some PWRs) or are movable (most PWRs).

Roughly one-third of the fuel assemblies in a PWR core are instrumented. For example, a three-loop Westinghouse PWR (157 assemblies in the core) has 50 instrumented core locations, and a four-loop PWR (193 assemblies in the core) has 58 such locations. Figure 4.1 illustrates the randomly distributed detector locations in a three-loop Westinghouse PWR.

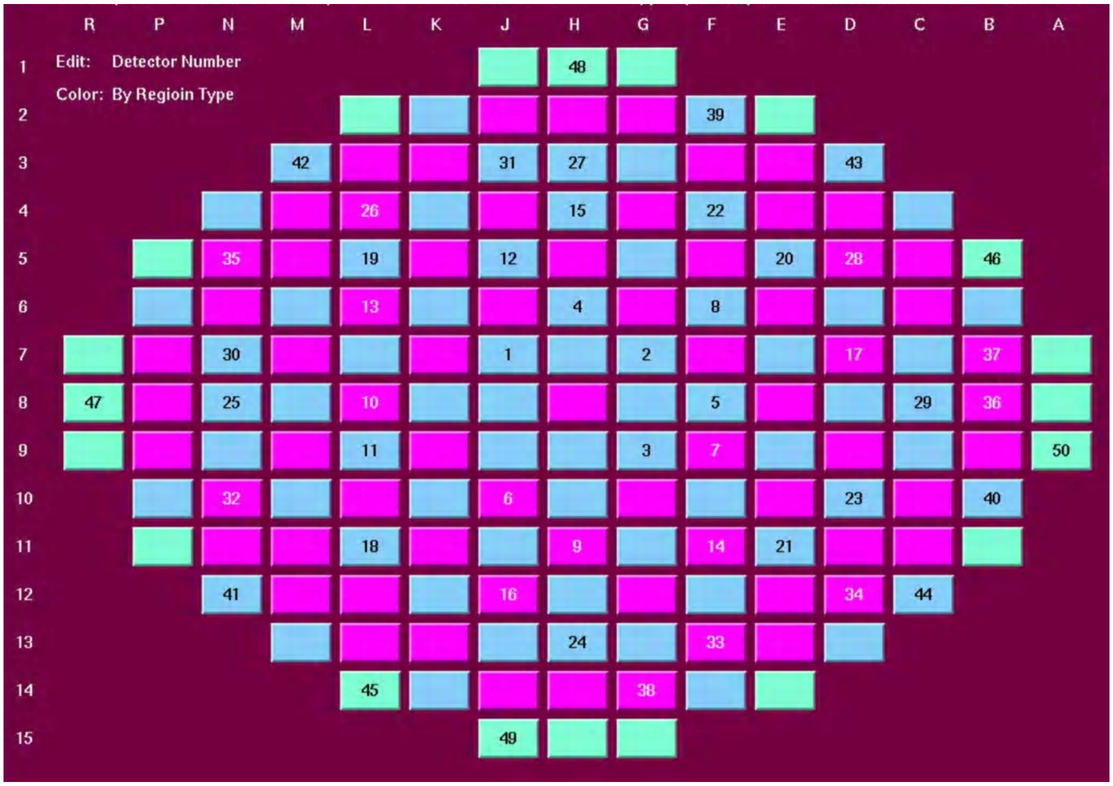

Fig. 4.1. In-core instrumented locations in a three-loop Westinghouse pressurized water reactor (Courtesy of Progress Energy).

A typical PWR movable in-core instrumentation configuration is illustrated in Fig. 4.2. Other reactor types have fairly similar systems. In a movable detection system, five or six fission chamber detectors are inserted through the instrumented core locations using retractable thimbles. The thimbles are closed at the leading reactor ends; thus they are dry inside and serve as the pressure barrier between the reactor coolant pressure and the atmosphere. Axial flux traces are recorded as the detectors are driven through the core. These traces typically contain 61 axial data points and provide detailed axial neutron flux variation along the fuel assembly. During the flux map measurement, which typically lasts several hours, it is imperative that the reactor conditions (power, temperature, control rod position, etc.) are held as constant as possible by plant operators: the goal is to obtain a core power distribution that is instantaneous (i.e., a snapshot of the reactor state, although several hours would have elapsed between the first and last trace measurements). The pertinent reactor conditions are recorded along with the detector signals for post-processing. For data normalization purposes, cross calibration of the detectors is performed by inserting all of the fission chambers through a common core location.

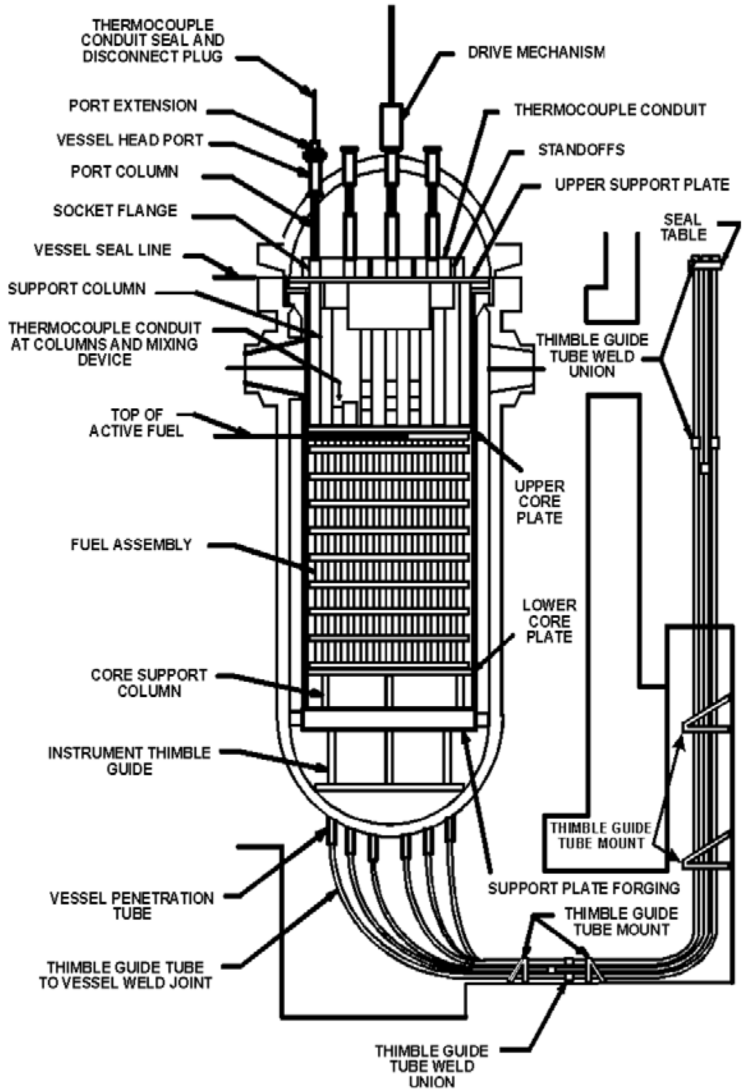

Fig. 4.2. Typical Westinghouse pressurized water reactor in-core flux map system.

4.4 IN-CORE MEASUREMENTS PROCESSING

4.4.1 Flux Map Processors

Even if all the fuel assemblies in the core were instrumented, the measured flux traces would still have to be processed. This processing step includes data validation checks, corrections for background and reactor operation fluctuations, evaluations of duplicate and symmetrical traces, and normalization to a single detector. As discussed previously, typically two-thirds of the fuel assemblies in the core are not instrumented, and power distribution at these locations needs to be determined. Several codes are available from the nuclear vendors to process the raw flux map data and generate the power distributions at the noninstrumented locations.[35,36,37] Utilities, through EPRI, have developed a core analysis system (ARMP) to support the operational, fuel management, and SNM accountability needs of their operational reactors. The ARMP system, which is typical of all such operational systems, has the ability to interpret and use in-core measurements to provide flux and power distribution data and to reflect changes in fuel and nuclear absorber characteristics as fissionable and absorbing materials are consumed.

Modern commercial flux map processors use proprietary methodologies that are relatively simple in concept and share the same general calculation flow. They are based on the use of predictions from the nodal simulator code and coupling factors relating noninstrumented locations to the nearest instrumented assemblies. The flux map processors significantly accelerate a process that in the 1960s and 1970s relied partially on hand calculation. Consequently, these processors eliminate the human factor element; ensure faster, more straightforward independent verification; and provide more easily retrievable core-follow records. Typically, the processors implement the following steps.

- The "raw" measured traces are checked for validity, corrected for background interference, statistically evaluated for any available duplicate and symmetric traces, and normalized to a single detector response.

- The assembly flux and/or power predictions are expanded from the nodal simulator number of axial nodes (~ 24) to that of the measured axial locations (~61).

- Measured reaction rates at the instrumented locations are compared to their predictions from the nodal simulator code.

- Correction factors are generated from the ratios of the measured to predicted reaction rates at the instrumented locations and are used to adjust the predicted assembly power to infer the power in the uninstrumented assemblies.

- Coupling or weighing coefficients for each noninstrumented location are derived using data from the nearest instrumented assemblies.

- The power distributions at the noninstrumented locations are derived using the nodal simulator assembly powers along with the coupling coefficients.

- Safety factors and margins to the Technical Specification thermal limits are computed using the inferred core power distribution.

Equation (5) is then used to determine the cumulative burnup from the inferred core power distribution and the cumulative reactor energy generation.

Figure 4.3 is an actual plant graph that illustrates the flux map processor comparative function. The comparison between the predicted and measured axial power distributions in an instrumented assembly is typical. In this figure the discontinuities in the measured data correspond to the presence of assembly spacers (which depress the flux) and correspond to the higher observed differences. For this particular case, the top and bottom of the assembly are blanket regions consisting of natural uranium used to increase efficiency, a feature of low-leakage core design. These blanket regions typically have higher differences because they are low power areas with high measurement statistical fluctuations. Outside of the spacers and blanket areas, the differences between predictions and measurements are in good agreement (generally below 5%). The purpose of the flux map is not only to validate the core-follow code but to determine the correction factors used to tune the predictions to yield accurate power distributions, and thus accurate thermal limits margins.

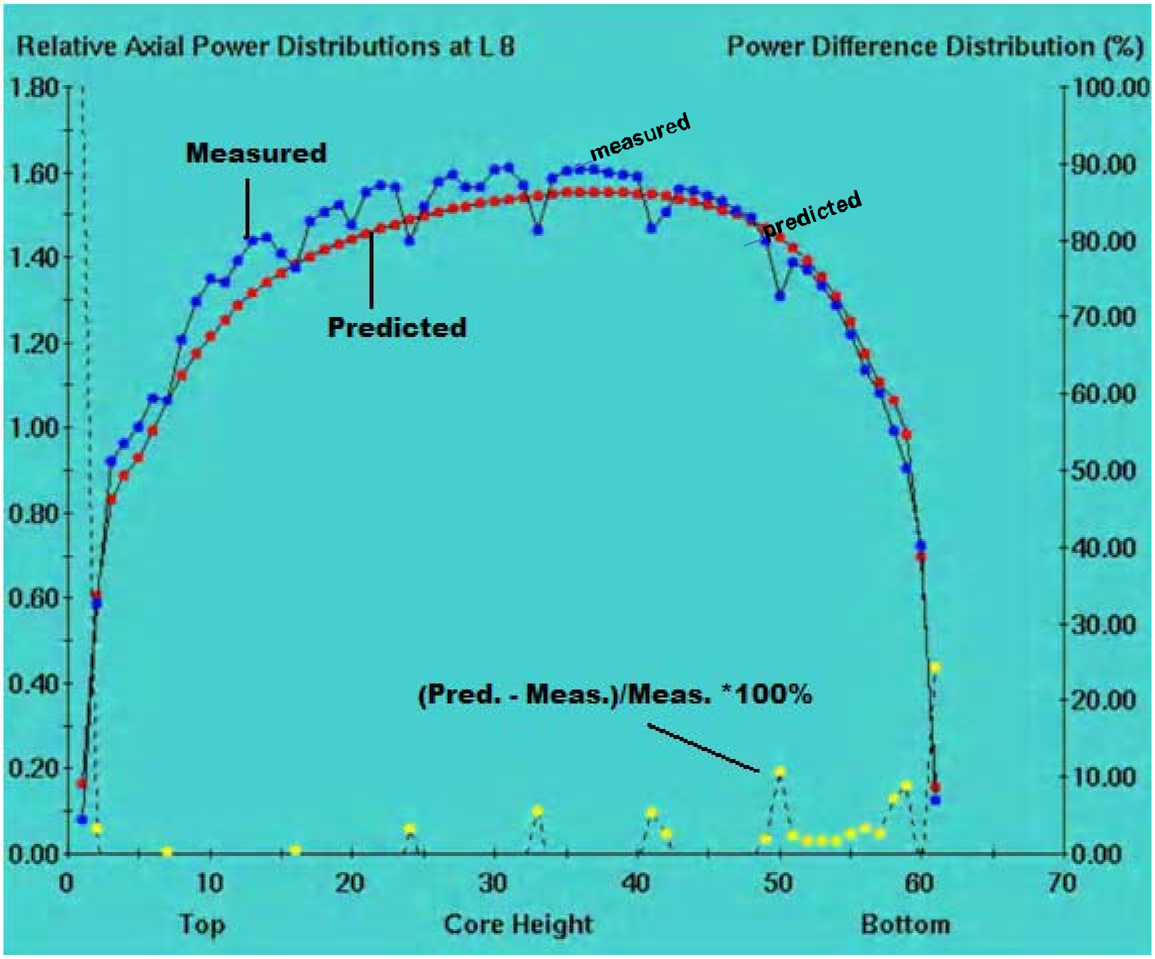

Fig. 4.3. Comparison of measured vs predicted axial power distribution at an instrumented core location (actual flux map data). The yellow points are the percent difference between the predicted and measured values, with negative differences not shown (Courtesy of Progress Energy).

4.4.2 Reactor Analysis Codes

The current methodology used to generate the core-follow data is approved by NRC for this purpose and is essentially the same that is used for reactor core design, augmented with in-core measurement processor codes. The most important among the several computer codes that constitute the design methodology are a lattice physics code (such as CASMO[38]) for fuel assembly modeling and a nodal reactor simulator code (such as SIMULATE[33]). The lattice physics code is a multigroup two-dimensional transport theory code used to conduct burnup calculations on an assembly or on a single fuel pin and handles a geometry consisting of cylindrical fuel rods with varying composition in a square-pitch array. Once the cross sections and other neutron transport constants are determined, they are used as input to the nodal simulator code. The nodal simulator code is a two-group three-dimensional program that solves the neutron diffusion equation for the homogenized nodal neutron flux. Typically, the reactor physics parameters are determined at 24 axial and 4 radial nodes in each assembly; this ensures detailed modeling of the core isotopic inventory and power distribution. During the core design phase, the nodal simulator depletes the core for the duration of the cycle at increments of about 1,000 MWd/MTU core average burnup and generates reactor parameters on a core, assembly, and node basis at each burnup. Using the cross sections and other physics constants for the major nuclides that are supplied by the lattice code, the

SIMULATE code updates the isotopic inventory at each reactor burnup. This inventory is determined at each core node and includes the atom densities that are necessary to determine the reaction rates, and subsequently the nodal flux and power. The output from the reactor physics code includes safety peaking factors (such as F_Q and $F_{\Delta H}$ for PWRs and minimum critical power ratio for BWRs), nodal power distribution, fuel assembly burnup, xenon and samarium reactivity worth, critical boron concentration, and core k_{eff}.

The typical nodal code also has a detector model that calculates the reaction rates as measured by a detector inside the fuel assembly during the flux mapping. This is an important validation feature that confirms the acceptability of the assembly axial distribution predictions.

During the core design process, reactor state points, such as 100% rated power and known rod positions (all control rods out of the core for PWRs, selected design rod patterns for BWRs) are modeled. The core-follow calculations involve depleting the nodal simulator to the current day and time in the cycle with the actual operating core state parameters. This ensures that the reactivity and core physics parameters are adjusted on a timely basis—daily, or hourly, depending on the utility—for the deviations between the assumed operating conditions in the design phase and the actual conditions.

It should be noted that use of microscopic depletion techniques to track individual actinide and fission product inventories in three-dimensional core simulator codes is a relatively recent development. Much of the current and historical fuel assembly burnup predictions and core follow data are not based on such a method.

4.4.3 Regular Validation of Reactor Analysis Codes

The primary driving force for accuracy in fuel assembly burnup measurement is derived from the regulatory-based requirement for accurate reactivity balances, as measured by the difference between actual and predicted control rod position (both BWRs and PWRs) or boron concentrations (PWRs). Accuracy in reactivity balances requires accuracy in power distributions, from which the burnup increments are directly determined. Thus, the level of accuracy of individual assembly burnup values is driven by the regulatory-based accuracy needs in reactivity balances.[31] Utility Technical Specifications require that the reactor reactivity be predicted within 1,000 pcm $\Delta k/k$. This is especially important for ensuring that the SDM, a calculated parameter that varies with cycle exposure, is accurately and conservatively determined. SDM is the ability to instantly bring down the reactor from full-power operation to shutdown by rapid insertion of the control rods. Generally, the utilities have a more restrictive acceptance criterion associated with reactivity prediction and typically use 500 pcm $\Delta k/k$. This criterion is verified regularly with reactor measurements: estimated critical position or estimated critical boron concentration, control rod position and daily updated boron letdown curve (required boron concentration at 100% rated thermal power).

Several of the parameters generated by the nodal simulator codes during the core design phase are used as inputs to the safety analyses codes and need to be confirmed by plant measurements before new cycle startup. The startup testing scope differs among the different reactor types because of the inherent design differences. For PWRs, the scope of this testing and the associated acceptance criteria are defined in an ANSI/American Nuclear Society (ANS) standard.[39] The PWR parameters that must be compared to in-core measurements during the startup testing, also called the zero power physics testing (ZPPT), include estimated critical conditions, core reactivity, control rod worth, and moderator temperature coefficient. It is clear that a successful verification of the ZPPT acceptance criteria is possible only if the three-dimensional nodal core power distribution is accurately predicted. For example, during one test, each control rod bank reactivity worth is measured at every few steps along the fuel assembly as the rod is withdrawn from the all-rod-in to the all-rod-out position. Each predicted reactivity worth along the 20 or so axial positions during the rod withdrawal must agree with the measured values consistent with the acceptance criteria in Ref. 39. Therefore, in addition to the total integral rod reactivity worth (expressed in percent mille), the predicted differential rod worth (percent mille/step) as a function of assembly height is validated.

Following successful ZPPT acceptance criteria confirmation, the reactor is allowed to increase power to a holding point—typically 30% rated thermal power—and the results of the ZPPT are saved as reactor records.

After completion of the ZPPT, a series of reactor measurements are taken at increasing reactor power levels up to full rated power as part of the power ascension testing. At about 30% rated thermal power, the first flux map of the cycle is performed. As noted previously, typically two-thirds of the reactor is composed of reinserted bundles whose burnup values were determined from the previous end-of-cycle flux map. One of the goals of this power ascension testing is to verify that each assembly is loaded in the correct position and has the burnup that was assigned to it from the last flux map. Should an assembly be misloaded or the burnup be significantly miscalculated, anomalous behavior should be noted in the reactor controls system, and per plant Technical Specifications, the situation would be evaluated and assemblies would be moved as necessary. Thus the startup physics testing provides a system to verify that the burnup of each assembly is consistent with the values used in the core design.

Note that the startup physics testing would not detect small errors in burnup close to fluctuation levels but might identify burnup errors of approximately 5% or more that would impact assembly reactivity. An actual example of a PWR core that was loaded with a number of assemblies containing reactivity different from the expected reactivity was documented in NRC Licensee Event Report (LER) 26193020. In this instance, due to an initial manufacturing deficiency, six new fuel assemblies were made with three fuel rods containing integrated burnable neutron absorbers placed in the wrong assembly quadrant. For this 17 × 17 assembly array, the three mislocated rods represent approximately 1% of the total number (264) of fuel rods. The uranium enrichment and composition of these rods were similar to those of the other rods except for the small amount (6%) of imbedded gadolinium compound. The core was loaded, and the reactor startup commenced. Upon completion of the first flux map at 30% reactor power, the utility engineers noted a radial core power tilt of 2.8%, which exceeded the acceptance criteria of 2%. The flux map indicated that the peaking factors were higher than expected, although still within Technical Specifications. The flux map program also produced a comparison of predicted vs measured fuel bundle relative powers, indicating a 14% higher power in the core area surrounding the misfabricated assemblies. The fabrication error was subsequently identified/confirmed via a review by the vendor of the fuel fabrication sheets. This example illustrates the sensitivity of the in-core detection system to small bundle reactivity differences between the design and actual core loading.

Once the first full-power flux map is successfully completed, the reactor is allowed to operate at rated full power. Monthly flux maps are then taken to confirm the core design power distribution predictions and to adjust core power peaking factors (e.g., $F_{\Delta H}$ and F_Q) to include measured deviations (e.g., core quadrant tilt & assembly radial and axial power variations) from design for comparison with safety limits to accurately determine thermal limits margins. Flux maps are also used to calibrate the ex-core detectors and determine additional reactor follow information such as core axial offset, a measure of the relationship between the amount of power generated in the top and bottom halves of the core, and quadrant power tilt ratio.

In addition to in-core measurements, the utilities evaluate the accuracy of their core power distributions and assembly burnup predictions by how efficiently they planned the reactor energy requirements and refueling outage schedules. It is important to compute the core loading requirements and use the associated energy to its fullest as fuel cost is by far the single largest expense for the upcoming cycle. Because a large fraction of the new core is composed of reinserted fuel assemblies whose burnup values are input to the reload design and safety analysis of the upcoming cycle, it is important for these values to be accurate. Also, plant outages are periods of reduced generating capacity, and the utilities schedule the refueling periods years in advance to ensure that only one unit is shutdown at any given time. Falling outside the planned reactor operating window can have significant cost impact, both in terms of generating capacity shortfall and failure to maximize the use of the fuel energy. Underestimating fuel burnup could mean operating at less than full power at the end of cycle and possibly starting the refueling outage earlier than scheduled; overestimation could mean fuel assemblies will be prematurely discharged without the full utilization of their energy. In both cases—that is, falling outside of the operating window assumed in the cycle-specific safety analyses—the utility is required to again perform these analyses under an accelerated schedule and at a significant cost. It is therefore important for economic reasons that the fuel assembly burnup values be accurately determined by the nuclear utilities. Overall, the industry has an excellent record of accurately predicting the cycle end of full power operations and meeting the refueling schedules start-end windows.

Successful reactor physics testing and good cycle length predictions give some unquantified confidence in the fuel assembly burnup values used in the reactor reload design process. However, it is not clear how far individual assembly burnup values would have to be in error in order to affect reactor performance, given the large number of fresh fuel assemblies present in the core at startup.

4.5 IN-CORE BURNUP MEASUREMENT ACCURACY

As discussed previously, the plant fuel assembly burnup record is technically not measured but inferred from measurements and code predictions. Approximately one-third of a core is directly measured, and the records do not generally differentiate between the data for instrumented and noninstrumented assemblies. Although a majority of the assemblies would be in an instrumented location for some portion of their reactor residency, very few would be in such a location in every cycle of their residency. However, there is a very large database of pairs of measurements and predictions at the instrumented locations because they are generated monthly during flux mapping. Comparisons of measurements to predictions are performed to quantify the accuracy with which the nodal simulator code calculates the power distribution and, thus, the burnup of a given instrumented assembly. A flux map summary report contains a detailed statistical evaluation of the measurements, predictions, and inferred data. Before methodology approval, vendors provide NRC with proprietary licensing topical reports [35,36,37] documenting the uncertainty associated with the flux map processor inferred power distribution based on actual plant data spanning several operating cycles. Consequently, most utilities have an acceptance criterion of about 5% between the observed measurements and predictions.

Previous studies [4,32,34,40] have shown that utility-developed burnup data are accurate to within at least 5%. A 1989 study, *The Incentives and Feasibility for Direct Measurement of Spent Nuclear Fuel Characteristics in the Federal Waste Management System*,[4] compared the accuracy of waste characteristics developed from utility-supplied data with the accuracy obtainable from direct measurements (in-core monitors). The report stated that “with respect to utility-supplied data it was concluded that because utilities measure total core power to within about 1% and make extensive in-core measurements of power distribution, assembly-average burnup values at end of life can be determined to within about 2% (to one standard deviation) of actual values.” In the document, *Reactor Record Uncertainty Determination,*[40] developed by the U.S. Department of Energy Yucca Mountain Project in 2004, the reactor records at nine PWR plants, consisting of 5,447 assemblies with assembly lifetime burnup values greater than 10 GWd/MTU, were evaluated, and their reactor records indicated an uncertainty in burnup of between 2 to 4.2% with a 95/95 confidence level.

In the EPRI report, *Determination of the Accuracy of Utility Spent Fuel Burnup Records*,[32] an evaluation of the core-follow code SIMULATE at only the instrumented locations was performed. Actual flux maps from three cycles were accessed, and only the measured reaction rates at the instrumented locations were extracted. Normally in the processing of a flux map, the subsequent steps of correction factors and/or coupling coefficients generation followed by the use of the core-follow code predictions at the noninstrumented locations would generate inferred values at these locations that collect additional uncertainty at each mathematical correction. Therefore, the EPRI study removes these additional "interferences" and evaluates the code predictions in only the locations with in-core measurements. Post-processing of the pertinent flux map data was performed using spreadsheets. After converting the measured reactions rates to measured assembly power, assembly burnup values were generated using the relationship between burnup and power [Eq. (5)]. The measured and predicted assembly powers were added and cumulated into measured and predicted burnup values at each time step, corresponding to the elapsed time between two successive flux maps. The uncertainty in assembly average burnup was expressed as the percent difference between measured and calculated burnup at each instrumented fuel assembly at the end of each of the three cycles. In the EPRI analysis, the area of concern was the over-prediction of burnup (which would be nonconservative). Therefore, a one-sided tolerance provided the appropriate level of probability and confidence. The tolerance was applied to ensure that 95% of the measurements occurred on one side of the normal distribution. Because the distribution was assumed to be normal, the 95% one-sided tolerance could be defined to include 95% of the error distribution. The resulting uncertainty for the assembly average burnup in the instrumented locations is consistent among the three cycles: 2.49%, 1.67% and 1.99% for cycles 1, 2, and 3, respectively.

The EPRI report also looked at movable detector reproducibility as that was a measure of uncertainty associated with repeated measurements of a signal detector in a given location. For each in-core flux map performed, a number of duplicate traces were obtained for reproducibility purposes at a given thimble location at different times during the flux map. In the three cycles of data analyzed, there were 441 pairs of duplicate traces. The mean error developed from this analysis was −0.147% with a standard deviation of 0.748%.

Utilities have periodically conducted burnup evaluation studies for design purposes or in response to external or regulatory requests. In a Duke Energy Corporation response to NRC concerning a Technical Specification amendment for the McGuire 1 and 2 spent fuel pool, Duke discussed a statistical evaluation of its assembly burnup records. To quantify the bounding burnup measurement uncertainty, data were compiled for the discharged fuel assemblies from McGuire's entire reactor operational history (≈ 1,900 assemblies from both units). When the code predicted and in-core measured burnup data were compared, the maximum individual assembly error (burnup measurement error) observed was about 4.0%.[34] Duke also noted that the measurement uncertainty showed a clear trend of smaller errors with higher discharge burnup.

In 2004, the Tennessee Valley Authority (TVA) evaluated its reactor records for burnup accuracy at all three of its reactor sites. To validate the fuel database burnup values, TVA performed a comparison of its flux-map-generated burnup values for discharged fuel assemblies against the design code burnup values. This analysis was done to identify any transcription errors, data entry errors, or other input errors in the database. Unless significant burnup value differences were identified (≥ 2% difference), new burnup calculations were not conducted for these assemblies. The comparison revealed differences between the predicted and database burnup values that were typically less than a few tenths of a gigawatt-day per metric ton of uranium and were randomly distributed. A total of 1,117 fuel assemblies were reviewed. Most burnup errors were less than 1 GWd/MTU, with the largest deviation observed from design values being 1.51 GWd/MTU on an assembly with approximately 41.8 GWd/MTU burnup (3.6% difference). Only seven assemblies met TVA's criteria for burnup value revision and required adjustments to their assigned reactor record burnup values.

Utility reactor records generally contain core and assembly-specific information that includes assembly design, burnup, initial assembly enrichment, power history, boron concentration (parts per million), moderator temperature, and discharge date. Record keeping requirements at individual utilities, for both the content and storage media of reactor records, have varied over time, as have the requirements for information required by NRC. For very old SNF, the reactor records may include "batch averages" for burnup information. The use of batch average values arose because the utilities were allowed to group assemblies with similar characteristics (a batch) into a single record for the purposes of reporting fuel inventories to NRC. As was noted in the report *Source of Burnup Values for Commercial Spent Nuclear Fuel Assemblies,* Rev. 3,[41] scoping examination of utility records for Three Mile Island Unit 1 (TMI-1) for plant core-follow burnup records produced mixed results.[42] This examination involved 12 discharged reloads of fuel, representing approximately 740 assemblies. About two-thirds of the reported assembly burnup values agreed well with the measured burnup values. About one-third of the reported burnup values reflected batch average values rather than individual assembly burnup values, as seen in Fig. 4.4.[43] The differences were generally between +5 and −5 GWd/MTU, with maximum differences upward of −16 GWd/MTU for some assemblies.[42] With regard to burnup calculations, the differences noted in the TMI-1 batch average data are not conservative and are cause for concern. Although not confirmed, it is expected that utilities have sufficient information to determine the individual burnup values for these assemblies currently reported with batch average values.

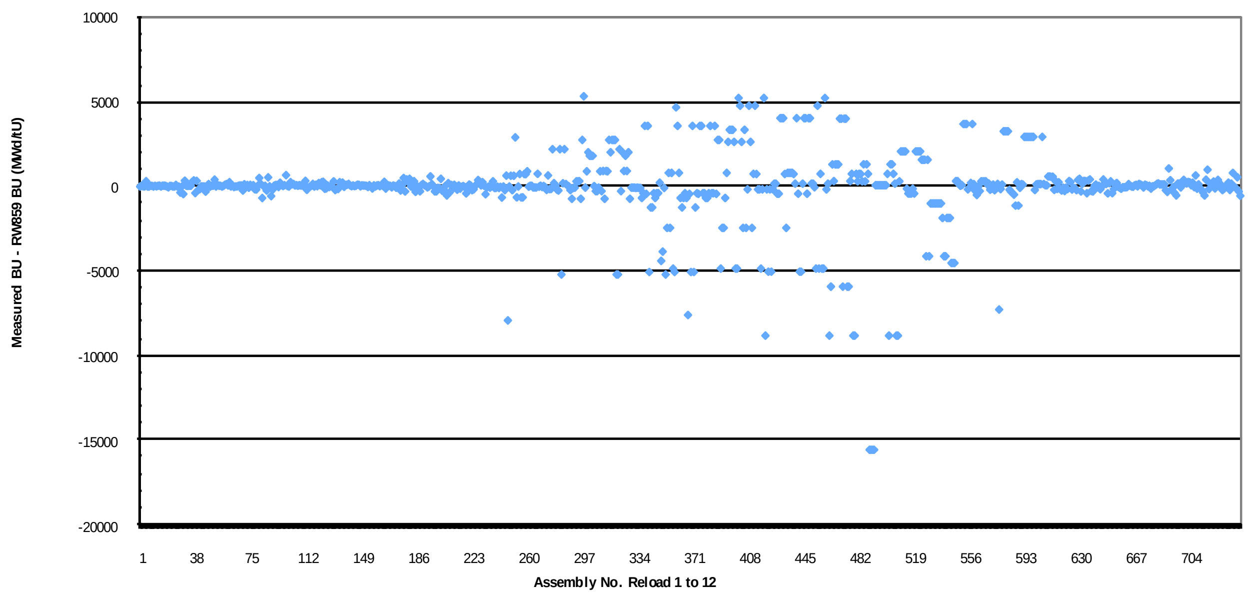

Fig. 4.4. Measured vs RW-859[44] reported assembly burnup (BU) values. (Source*:* Ref. 18.)

Two other factors affecting the accuracy of fuel burnup calculations are the use of increased irradiation times in reactors (increased fuel burnup) and the installation in many reactors of more precise feedwater flowmeters. Increasing the burnup has the effect of "equalizing" burnup uncertainty. As was noted earlier, in the EPRI report, *Determination of the Accuracy of Utility Spent Fuel Burnup Records*,[32] an analysis was conducted on discharged fuel from three cycles of operation. This covered a startup cycle, a transition cycle, and a near-equilibrium cycle. At the end of each cycle, about one-third of the fuel assemblies in the core were discharged. Since only about 30% of the core locations were instrumented, this meant that about 34% of the assemblies were never located in an instrumented location. Of the remaining assemblies, 44% were instrumented during one cycle, 19% were instrumented during two of the cycles, and 3% were instrumented in all three cycles. The uncertainty in burnup evaluated over three cycles of operation demonstrated a general decrease in uncertainty with an increase in residence time or burnup. For assemblies discharged after one cycle of burnup, the mean burnup uncertainty was 1.9%; after two cycles of burnup, the mean burnup uncertainty was 0.98%; and after three cycles of burnup, the mean burnup uncertainty was 1.02%. The data were not weighted for different cycle lengths, and all data were given equal weight. This decrease in uncertainty is indicative of the self-correcting nature of burnup, which is seen as the decrease in assembly average burnup uncertainty as the in-core residence time of assemblies increases.[32] The use of more precise feedwater flowmeters, which were approved for use in the 1980s, reduces the uncertainty in determining feedwater flow. The feedwater flow is a direct input to the calculated reactor thermal power, which is used in core-follow fuel burnup calculations. Thus since the advent of the new flowmeters, reactor thermal power can be calculated with less uncertainty, and the resultant assembly core-follow calculations for burnup can be more accurately calculated.

Another way to evaluate the accuracy of total core burnup measurements would be to look at the change in boron concentrations in the core throughout a fuel cycle. As noted previously, the differences between true burnup values and measured burnup values in the entire core are expected to be small. In a nominal PWR core, at the start of a fuel cycle, boron concentrations are about 1,600 ppm and steadily decrease as fissile content in the fuel is depleted. By the end of a 15-month fuel cycle (16,000 MWd/MTU), the boron concentration in the core has been reduced to nearly zero—which is a decrease of 3.5 ppm per day. If the total core burnup were miscalculated by 5%, this would equate roughly to missing the end-of-cycle full-power capability by more than 22 days, which would be very obvious. By evaluating the boron concentration in the core, a utility can show that integral core power, and thus true core burnup, over any fuel cycle has been calculated accurately.

One factor that impacts the uncertainties in assembly burnup calculations is the evolution of the computer codes used to calculate core power measurements. In the 1970s and 1980s, SNF assembly burnup values were inferred from monthly in-core flux measurements using cycle-specific data, typically supplied by the reactor vendors. Due to computing resource limitations in those days, these constants were prepared using two-dimensional fine-mesh (pin-by-pin) models. The methods used were adequate for the original purposes of the in-core flux measurements, which were primarily to identify misloaded assemblies, dropped control rod clusters, and cross-core power tilts. Since that time, however, there have been significant improvements in computing capabilities and in the software used to model reactor cores to provide continuous real-time monitoring capability. Burnup credit has been used for storage of commercial SNF at nuclear power plants since the 1980s as utilities worked to increase their on-site storage capacity. Reliance on the plant-supplied assembly burnup values for burnup credit has raised awareness of the importance of determining accurate burnup values. An important factor responsible for the improved accuracy of assembly burnup values is the improvement in quality assurance (QA) programs throughout the nuclear power industry, encouraged by NRC. Improved computing resources and methods, use of continuous real-time core monitoring systems, increased awareness of the need for accurate assembly burnup data, and significant improvements in QA practices have resulted in improved reliability of the assembly burnup data.

Factors that could negatively impact the accuracy of reactor burnup records include human error, transcription errors, and degradation of information storage media. Since the 1980s, utilities have maintained burnup records for individual assemblies using flux map processor data. While these data may be as accurate as noted in the reports above, the assembly burnup data have often been transcribed (possibly several times), input into new SNF tracking software, and/or stored on non-QA-audited media. Transcription errors for the burnup data or assembly identifiers or legibility issues with the media could result in “incorrect” information being used in cask loading operations.

4.6 SUMMARY

The burnup values assigned to fuel assemblies by the utilities are not measured. They are inferred from neutron flux measurements and from core thermal power, which is itself inferred from a reactor energy balance. Uncertainties associated with determining burnup values include (1) the uncertainty in the flux measurements, (2) the uncertainty in the conversion of the flux to assembly power, (3) uncertainty in the extrapolation and/or interpolation of assembly power to noninstrumented assemblies, (4) uncertainty in the determination of the core thermal power, and (5) uncertainty in the integration over time of assembly power to generate burnup values. Largely qualitative comparisons of the assembly-specific inferred burnup values, measured reactor physics test results, and core follow data with design predictions give some confidence that the inferred burnup values are close to the true burnup values. Quantitative comparisons showing the sensitivity of physics test results and core follow data to variation in assembly burnup would be helpful in bounding fuel assembly burnup uncertainty.

Accurate determination of the assembly burnup, the amount of cumulative generated power per mass of fuel, depends on an accurate determination of core power distribution and core thermal power output history. Although the core power distribution that is predicted by validated and approved core design codes is acceptable, in-core reactor measurements are used on a regular basis to update these predictions to reflect the actual reactor operating conditions. Commercial flux map processing codes claim high accuracy for the resulting power distribution, and utility reactor records appear to be accurate to within 5%. Multiple independent comparisons performed at different utilities between 1985 and 2004 involving several thousand in-core measured assembly burnup values report reactor record deviations of less than 4.2% from core design predictions. A notable exception involves records from some utilities that report batch average burnup values. Also, the industry records do not take into account any inaccuracies in recording or transcribing the assembly burnup data into utility databases. Finally, note that there is some expectation that the accuracy of the reactor record values is fairly constant over the years due to competing effects; computer codes have become more accurate over time, tending to improve accuracy, while fuel designs and, to some extent, operations have become more complex, tending to reduce accuracy.

5 CONSEQUENCES OF FUEL ASSEMBLY MISLOADING

As discussed in Section 1, a loading curve (see Fig. 1.1) represents combinations of minimum burnup and initial enrichment that correspond to a limiting value of the effective neutron multiplication factor (k_{eff}) for a given configuration. Assemblies with insufficient burnup, as compared with the minimum required burnup dictated by the loading curve, are not acceptable for loading, and thus if loaded, would be considered to be "misloaded." The goal of the NRC recommendation in ISG-8 for verifying reactor records by out-of-core measurement is to prevent inadvertent loading (i.e., misloading) of underburned assemblies. To understand the significance, and corresponding diligence with which such misloadings should be prevented, it is important to understand the consequences of potential assembly misloading on the system k_{eff} value and to evaluate the associated increases in k_{eff} against inherent margins (e.g., an administrative margin), where present. To support this understanding, a study[45] was recently performed to determine the changes in k_{eff} that can result from a wide variety of postulated fuel misloading events in a representative high-capacity transport cask. The purpose of the study was to provide quantitative information for the impact of such events on the subcritical margin and aid in assessing the appropriate role for burnup measurements in ensuring safety. The discussion that follows summarizes the analysis and findings in Ref. 45.

Misloading an underburned fuel assembly will cause an increase in the cask reactivity. The exact magnitude of the increase is dependent on several factors, including cask design, assembly design, assembly irradiation conditions, post-irradiation cooling time, assembly enrichment, assembly burnup, and assembly position within the cask. However, the magnitude of the increase is dominated by the amount by which the actual assembly burnup is less than the minimum required burnup value for loading acceptance and the position of the assembly within the cask. Therefore, the computational study was performed to investigate a variety of cases involving misloading of underburned assemblies, including cases involving misloading of assemblies with no burnup (i.e., with fresh fuel), into the center positions of the generic 32-PWR assembly (GBC-32) cask design. The GBC-32 was developed to be representative of typical high-capacity rail casks being considered by industry. Detailed specifications for the GBC-32 cask are provided in *Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit*.[46] In all cases considered for the study, the GBC-32 cask was assumed to be fully flooded with full-density water, and for all assemblies in all cases, Westinghouse 17 × 17 fuel and a cooling period of 5 years were assumed. To provide a bounding estimate of the effect of misloading an underburned assembly, the misloaded assemblies were assumed to be in the central positions for all cases. For all other potentially dependent factors, representative conditions were assumed (i.e., a representative high-capacity cask design, representative burnup credit depletion conditions, representative fuel assembly specifications, and a 5-year cooling time).[45]

The investigation of misloading underburned assemblies was performed to provide estimates of the effect of misloading events involving errors in assembly burnup verification, perhaps due to inaccurate reactor records, while the investigation of misloading fresh fuel with different enrichment values provides estimates for worst-case misloading events, potentially due to improper assembly identification. Although no attempt was made to quantify the probability of such misloading events, others[47] have estimated that the probability of a single misloading can be as high as 10^{-3} or as low as 10^{-5} for a large cask. If it is valid to assume that multiple misloading events in a single cask are statistically independent, the probability of misloading *n* assemblies can be estimated by raising the probability of a single misloading to the power *n*. Examination of known misloading events (see Section 6) reveals examples in which multiple misloads are not independent.

The study considers two representative burnup points on cask loading curves developed with and without the principal fission products present and includes an investigation of (1) the effect of misloading fuel that has burnup values that are 90, 80, 50, 25, 10, and 0% of the minimum value required by the loading curve (i.e., underburned); (2) the effect of misloading fresh fuel with enrichment values of 2, 3, 4, and 5 wt % ^{235}U; and (3) for each scenario, the effect of misloading multiple (1, 2, 3, and 4) assemblies. The actual burnup and enrichment combinations used in the study were 30 GWd/MTU, 3.00 wt % ^{235}U and 45 GWd/MTU, 3.66 wt % ^{235}U for conditions representing actinide-only burnup credit and analysis assumptions consistent with ISG-8 and 30 GWd/MTU, 3.88 wt % ^{235}U and 45 GWd/MTU, 4.89 wt % ^{235}U for conditions representing credit for actinides and the principal fission products.

For the representative burnup-enrichment combinations corresponding to actinide-only burnup credit, it was found that misloading one assembly with 0% of the required burnup (i.e., fresh fuel with enrichment of either 3.0 or 3.66 wt % ^{235}U) results in an increase in k_{eff} of 0.02–0.035. Likewise, misloading two assemblies that are underburned by 75% (i.e., have only 25% of the required minimum burnup) results in an increase in k_{eff} of 0.025–0.035, while misloading four assemblies that are underburned by 50% also results in an increase in k_{eff} of 0.025–0.035. Because the reactivity reduction due to burnup is increased with the inclusion of credit for fission products, the impact of underburned fuel is also larger compared with corresponding cases without fission products present (i.e., actinide only). Consequently, misloading one assembly with 0% of the required burnup (i.e., fresh fuel with enrichment of either 3.88 or 4.89 wt % ^{235}U) results in an increase in k_{eff} of 0.035–0.055 compared with 0.02–0.035 for the actinide-only case. Misloading two assemblies underburned by 75% or four assemblies underburned by 50% results in an increase in k_{eff} of 0.035–0.045. For all cases considered, four simultaneous misloads involving 20% reduced burnup results in a maximum increase in k_{eff} of 0.0125, which is well within the 0.05 administrative margin that is typically used in these types of criticality safety evaluations. Another notable observation is that for the cask and conditions considered, a reduction in burnup of 20% in all assemblies results in an increase in k_{eff} of less than 0.035. Note, in all cases considered, the "misloaded" assemblies were assumed to have reduced burnup; no cases were considered in which the burnup was increased as compared with the minimum required value. Also, in all cases, the remainder of the cask was assumed to be loaded with assemblies that just met the burnup-enrichment loading criteria (i.e., burnup and enrichment combination on the loading curve). Hence, the analysis made no attempt to credit the practical reality that a significant portion of the loaded assemblies may have higher-than-required burnup, which can compensate for the positive reactivity insertion associated with an underburned assembly.

For misload conditions involving fresh fuel assemblies, without regard to the burnup-enrichment combinations from a loading curve, the results indicate that misloading a single fresh assembly with 3, 4, or 5 wt % ^{235}U enrichment results in an increase in k_{eff} of approximately 2, 4, or 6%, respectively. Notably, a single fresh fuel assembly with 5.0 wt % ^{235}U enrichment loaded into the cask center will result in an increase in k_{eff} of more than the 0.05 administrative margin, while misloading two fresh 5 wt % ^{235}U assemblies results in an increase in k_{eff} of more than 0.10. Because the burnup-enrichment combinations correspond to the same value of k_{eff} and the misloaded fresh fuel assemblies are not dependent on the reference burnup-enrichment combination or the nuclides included in the spent fuel compositions, these results were not dependent upon the presence of fission products. Throughout the analysis, the misloaded assemblies were placed in the most reactive (central) positions within the cask. Therefore, the impact of misloading assemblies into noncentral positions (e.g., nearer to the radial cask periphery) is bounded by the cases considered.[45]

In summary, the consequences to k_{eff} of loading assemblies that have slightly reduced burnup (e.g., 5% due to uncertainties in the burnup verification process), as compared with the required burnup, are fairly small (≤ 1%). On the other hand, loading one or more highly enriched (i.e., > 4 wt %) fresh fuel assemblies has a significant consequence on criticality safety. These findings suggest that while it may not be overly important to precisely verify the burnup value, it is important to ensure that fresh or very-low-burnup (nearly fresh) fuel assemblies are not misloaded into a cask.

6 FUEL MOVEMENT AND MISLOADING EXPERIENCE

The recommendation in ISG-8 for administrative procedures to include a measurement that confirms the reactor record for each assembly is intended to prevent unauthorized loading (misloading) of assemblies due to inaccuracies in reactor burnup records and/or improper assembly identification, thereby ensuring that the appropriate subcritical margin is maintained. Section 5of this report detailed the effects a misload (or multiple misloads) could have on criticality. It is important to recognize that misloading of SNF, although rare, does occur.

Fuel-handling problems can be categorized into one of the following categories.

1. The desired (correct) assembly can be selected and moved, but it is moved to an incorrect location. This category has two subsets.

   a. It can occur through human error (placing the assembly in location B when it was intended for placement in location A).

   b. It can occur through procedural/personnel error. When the assembly is purposely placed in location A, later review of the Technical Specifications may reveal that the assembly does not meet the requirements for being in location A, and it should have been placed in location B. This error can occur if incorrect data are used to select the assembly (e.g., incorrect burnup, cooling period, or initial assembly enrichment information), incorrectly transcribed information from databases/calculations is used, incorrect procedures are used, or correct information/procedures are applied incorrectly.

2. Fuel can be misidentified (wrong assembly selected and then moved to the designated location).

Utilities implement a number of safeguard methods to prevent fuel misloads. These methods include hardware, procedures, and redundancy of assembly verification. Each fuel assembly is manufactured with a unique identification number (ID) permanently installed on the upper assembly plate that enables effective visual confirmation of each assembly ID (Figs. 6.1–6.2). Plant procedures require independent verifications (by the qualified person performing the work and a second qualified person) of the assembly ID before each fuel movement. When all the fuel bundles have been moved to their intended locations—either in the core, in the spent fuel pool, or in the spent fuel cask—independent verifications by another crew member are performed to compare the IDs of the as-loaded configuration to those in the map provided in the implementing design document.

To evaluate the frequency and causes of such misloading/misplacement incidents, NRC's Sequence Coding and Search System (SCSS) database for LERs was searched for incidents of misloaded fuel assemblies. Commercial nuclear plant licensees must submit LERs pursuant to 10 CFR 50.73 to document specific types of operational events, including (among others) operations or conditions prohibited by the plant's Technical Specifications, degraded conditions of the plant or its principal safety barriers, operation in an unanalyzed condition that degrades plant safety, or events or conditions that could have prevented the fulfillment of the function of structures or systems important to safety. Misloaded assemblies of SNF typically occur in the spent fuel pool and are reported as violations of Technical Specifications. Core misplacements, though rare, would be reported as operation in an unanalyzed condition. It should be noted that the database of LERs covers a limited period of spent fuel pool movement experience, and that it is not clear that all misload events that occurred during that period are accurately reflected in the database. Given that this is the case, it is difficult to draw quantitative conclusions about spent fuel pool misload frequency and trends.

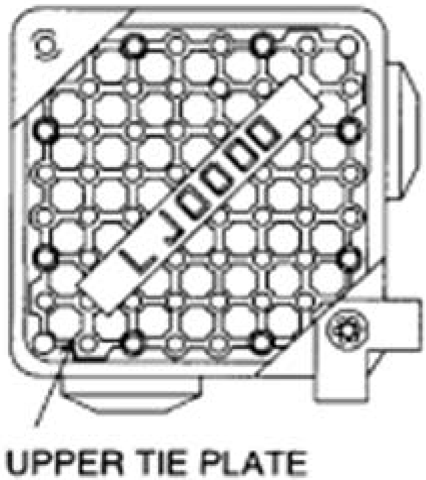

Fig. 6.1. Drawing of a boiling water reactor fuel assembly with assembly identification number.

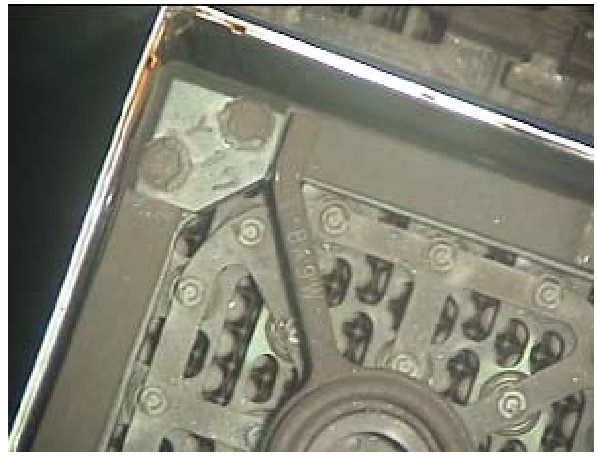

Fig. 6.2. Picture of the top of a pressurized water reactor fuel assembly in a spent fuel pool showing the unique assembly identification number.

The NRC's LER database was searched to find and retrieve LERs reporting fuel-handling errors and misloaded SNF assemblies from the approximately 50,000 LERs reported from 1980 to early 2003, when the SCSS was discontinued. All subsequent LERs were manually reviewed to find additional pertinent events in the 2003–2006 time period. A number of LERs were identified as potential misload/misplacement events. Upon review, it was found that in all but one event the fuel assemblies were selected and handled correctly (i.e., consistent with the intent). However, the assemblies were subsequently determined to have been loaded in locations in the spent fuel pool contrary to technical specification requirements as a result of calculational, procedural, or other human errors. Thus, the assemblies were mispositioned; they were not incorrectly identified or mishandled. Reactor records were responsible for two of the LERs. A total of 19 LERs that reported pertinent fuel-handling errors were located and are summarized below.[48]

- (LER 36682036) In 1982, Hatch Unit 2 personnel loaded the wrong fuel assembly around a source range monitor. Bridge personnel mistakenly picked up a bundle adjacent to the desired bundle in the spent fuel pool. Thus, the cause was personnel error.

- (LER 36683031) In 1983, Hatch Unit 2 personnel loaded eight fuel assemblies in an improper core orientation around a source range monitor. Plant personnel entered the fuel assembly data into the plant computer incorrectly by entering the data as Unit 1 data instead of Unit 2 data. Thus, the cause was personnel error.

- (LER 36985035) In 1985, McGuire Unit 1 personnel identified seven spent B&W fuel assemblies stored side by side in the spent fuel pool in a manner contrary to Technical Specification requirements. The seven assemblies did not meet the burnup criteria for unrestricted placement in Region 2 of the spent fuel pool and should have been stored in a checkerboard pattern. This occurred due to personnel error in not adequately reviewing the Technical Specifications before approving the fuel movement.

- (LER 26685005) In 1985, a Point Beach Unit 1 QA audit determined that three spent fuel assemblies that had been discharged from the reactor less than 1 year had been stored next to the spent fuel pool wall in violation of plant Technical Specifications.

- (LER 21987006) In 1987, Oyster Creek Unit 1 personnel temporarily stored 184 fresh fuel assemblies in the spent fuel pool that had enrichments above Technical Specification requirements. The safety analysis for the fresh fuel was prepared based on the assumption that the fuel would be stored only in the dry storage facility. The cause of the violation was personnel error. A Technical Specification modification was subsequently requested to allow storage of the more highly enriched fuel in the spent fuel pool.

- (LER 28687008) In 1987, Indian Point Unit 3 personnel identified a fresh fuel assembly that had been inadvertently loaded in the spent fuel pool in a manner contrary to Technical Specification requirements. The misloading was due to personnel error.

- (LER 30287026) In 1987, Crystal River Unit 3 personnel moved a fresh fuel assembly in the spent fuel pool in a manner contrary to Technical Specification requirements. The misloading of the assembly was promptly identified, and the assembly was relocated to an approved location. The misloading was due to personnel error.

- (LER 28088028) In 1988, Surry Unit 1 personnel found that they had placed a spent fuel assembly in a location in the spent fuel pool in a manner contrary to Technical Specification requirements. The spent fuel assembly had not decayed for a sufficiently long time period to be stored in the region in which it was placed. Subsequently all spent fuel in that region of the spent fuel pool was confirmed to meet plant Technical Specifications. The cause of the misloading was procedural errors.

- (LER 36991016) In 1991, McGuire Unit 1 personnel identified 11 fuel assemblies that had been stored in the Unit 1 spent fuel pool in a manner contrary to Technical Specification requirements. The vacant row provision of the Technical Specifications had not been met. The problem was determined to be due to a procedural deficiency because the guidance provided by the applicable procedure was obscure.

- (LER 45494006) In 1994, Byron Unit 1 personnel identified a fuel assembly located in the wrong region of the spent fuel pool. The assembly did not meet the Technical Specification requirement for burnup. The Nuclear Component Transfer List (NCTL) incorrectly specified the placement of the assembly into a region not approved for the actual assembly burnup. The cause was personnel error. A search by plant staff found one previous event of a misloaded fuel assembly due to an error in the NCTL. (This second event was not found in this LER database review but was noted in the subject LER by plant staff.)

- (LER 45696007) In 1996, Braidwood Unit 1 personnel placed spent fuel in an inappropriate configuration during a Boraflex neutron absorber testing procedure. Staff did not consider the effects of the assemblies on lower-burnup fuel in adjacent storage locations. The cause was procedural, personnel, and managerial error.

- (LER 45496008) In 1996, Byron Unit 1 personnel identified three fuel assemblies located in a region of the spent fuel pool in a manner contrary to Technical Specification requirements. The assemblies did not meet the Technical Specification requirements for minimum burnup. The computer spreadsheet used to verify minimum burnup contained erroneous information and had not been independently verified. Subsequently, all SNF assemblies in region 2 of the spent fuel pool were verified to be in their correct locations per Technical Specifications.

- (LER 45696008) In 1996, Braidwood Unit 1 personnel identified a spent fuel assembly located in a region of the spent fuel pool in a manner contrary to Technical Specification requirements. The cause was personnel error. The fuel assembly did not meet the required checkerboard configuration based on burnup vs initial enrichment specified in the Technical Specifications. A calculation contained the wrong burnup information and was not independently verified. In this LER, Braidwood staff noted one other fuel misloading event (457-200-94-016), in which new fuel was misloaded in the spent fuel pool during transfer from the new fuel storage vault. The cause of the event was personnel error in not following procedures.

- (LER 52999003) In 1999, Palo Verde Unit 2 personnel identified a spent fuel assembly stored in the wrong position in the spent fuel pool (based on the initial enrichment and assembly burnup factors). This misloading occurred due to personnel error and poor independent verification. A subsequent review of the spent fuel configuration of all Unit 1, 2, and 3 assemblies confirmed that out of approximately 2052 assemblies, only the one assembly was in the incorrect location.

- (LER 34800004) In 2000, Farley Unit 1 personnel identified three spent fuel assemblies in spent fuel pool locations not allowed by Technical Specifications. The assemblies had insufficient burnup for the storage locations in which they were placed. The cause was personnel error, lack of detail in the procedure, and insufficient independent review. Subsequently all Unit 2 spent fuel pool assemblies were confirmed to be in the correct locations per Technical Specifications.

- (LER 27502006) In 2002, Diablo Canyon Unit 1 personnel determined that a spent fuel assembly had been stored in a manner contrary to Technical Specification requirements. The assembly had insufficient burnup for its location adjacent to other spent fuel assemblies. The cause was personnel error. Subsequent to this event, the utility verified the spent fuel pool storage configuration for the previous 5 years and found no other discrepancies.

- (LER 24404002) In 2004, Ginna personnel identified three (old) consolidated rod storage canisters in spent fuel pool locations not allowed by Technical Specifications. A subsequent review indicated that in early storage calculations, the total energy generated by the fuel rods contained in the canisters was incorrectly calculated. Later, when the canisters were moved in the spent fuel pool, only the total canister data (for each canister) was reviewed, not the data on the individual assemblies in each canister. The cause was personnel error for using incorrect burnup values calculated for the fuel assemblies inside the three canisters.

- (LER 44504001) In 2004, while performing a complete verification of fuel assembly configurations in the spent fuel pool, Comanche Peak personnel identified one fuel assembly loaded in a manner contrary to Technical Specification requirements. The cause was an incorrect burnup value that had been calculated for the fuel assembly. The incorrect burnup value resulted from inadequate conversion of the data files during a computer code migration in 1998. During a subsequent verification of all assembly configurations in the spent fuel pool, several other discrepancies in fuel assemblies were identified. These discrepancies were compared against the applicable Technical Specifications, and it was concluded that all the affected fuel assemblies (other than the initial assembly) were acceptable in their existing storage locations.

- (LER 31106004) In 2006, while reviewing the Salem Unit 2 move sheet for the fifteenth refueling outage core reload, it was discovered that a fuel assembly had been placed in a region of the spent fuel pool without meeting the minimum burnup requirements for that storage location. During the subsequent investigation, it was discovered that 11 other fuel assemblies had also been inappropriately discharged into similar locations in the Unit 2 spent fuel pool without meeting the minimum burnup requirements, and 1 assembly had been inappropriately placed in the Salem Unit 1 spent fuel pool. The cause of the occurrences was attributed to a lack of technical rigor associated with the review and verification of assembly burnup values before discharge into the spent fuel pools; ultimately, personnel error.

Another misload event was identified in the NRC "Morning Reports." In November 2000, the utility Consumer's Power notified Region III that a total of 11 spent fuel assemblies had been placed into five VSC-24 spent fuel storage casks at the Palisades Plant before undergoing 5 years of cooling time (as required by the cask CoC) following their discharge from the reactor core. (They had cooled for just over 4 years.) The event was due to an error in recording the core discharge dates for these assemblies. This event was not treated as a criticality issue but was evaluated from a heat-load standpoint and does represent a misload event.

During the preparation of this report, a Certificate of Compliance Violation Report[49] related to the misloading of 34 fuel assemblies in four HI-STORM 100 model casks at Grand Gulf was filed by Entergy. The fuel assemblies "exceeded the maximum allowed Decay Heat per Fuel Storage Location Limit and/or the Fuel Burnup Limit." The misloading was preliminarily attributed to an error in the Cask Loader Database, which was used to select the fuel assemblies that were loaded into each cask. The Cask Loader Database was relied upon to ensure that assemblies selected for loading met the applicable exposure and cooling time limits provided in the CoC. Similar to the situation with the VSC-24 cask, this event does not represent a criticality concern, but does represent a misload event.

A review of the above-listed LERs illuminates a number of issues. Only one LER (36682036) deals with the misidentification of a spent fuel assembly. Of the other LERs, 16 LERs describe spent fuel pool misloadings of assemblies due to deficient procedures or human error in not following approved procedural requirements. Two LERs were related to a reactor record error. The details behind LER 24404002 (2004, Ginna) show that the spent fuel assemblies had been sent off-site as part of a testing program and were returned several years later to the site spent fuel pool. The SNF was stored in consolidated-rod storage canisters in the spent fuel pool. The site Nuclear Fuel Accounting Code contained canister data but not data for the individual fuel assemblies that were contained in the canisters, even though the site evidently did have individual assembly records on file.

Another important observation from this review is that there is very little data on misloads in spent fuel casks—of all the information reviewed, only two fuel assembly misloading events involved spent fuel casks. Hence, the applicability of the above misload data to cask loading operations is not entirely clear. The existence of relatively little data for cask misloading events is expected since fuel movements into spent fuel casks have been relatively few, as compared to fuel movements associated with the core and spent fuel pool.

Note that the review of the LER database for this report certainly did not find all SNF misloading events that have occurred at utilities. The two misloading events referenced in the LERs listed above and in the "Morning Reports" discussing the assembly-cooling-time issue are indications of other misloading events. Even so, the frequency of misloads of assemblies appears to be low. Per the 2002 RW-859 report,[44] as of the end of 2002, 163,646 SNF assemblies in the United States had been permanently discharged from reactors. The U.S. reactors discharged approximately 29,000 additional spent fuel assemblies between 2003 and 2006. It seems reasonable to assume that each assembly has been moved at least eight times (once into the spent fuel pool as fresh fuel, once into the core for cycle 1, shuffled two times each for cycles 2 and 3, once back into the spent fuel pool as burned fuel, and once within the spent fuel pool as the fuel ages and cools). A significant number of assemblies will have been moved more frequently, as spent fuel pool contents are rearranged due to heat load distribution issues, spent fuel pool reracking, space management realignments, support for outage needs, etc. Thus, a reasonable estimate of fuel assembly moves may range from about 8 moves per assembly to 10 or more moves per assembly. Considering the large number of assembly moves and the relatively small number of misloading events, the frequency of occurrence is rather small and given the numerous lessons learned, particularly in terms of procedural development and compliance, this frequency should decrease with time. However, more complex loading patterns and increases in the number and types of fuel assemblies present in a spent fuel pool appear to act against such a decrease in this frequency, making it difficult to draw quantitative conclusions about spent fuel pool misload frequency and trends in the absence of more data.

The LERs evaluated show that the probability of selecting an assembly with the wrong ID or placing the assembly in a position other than the position specified by procedure is very low (only one occurrence in the 19 LERs listed). The probability of the procedures being in error or of human error occurring in implementing the procedures, while also very low, was the predominant factor in the other assembly misloading events. Additionally, note that as corrective actions for many of these LERs, the affected utilities verified that the assemblies in the spent fuel pools were stored in their correct locations according to plant Technical Specifications. In so doing, because other fuel misloadings were not discovered, the utilities demonstrated that their spent and fresh fuel movement and storage programs did not have additional and/or systematic errors.

A major criticality concern when loading casks is the potential for misloading fresh or very low burnup (nearly fresh) fuel into positions in lieu of spent fuel assemblies. Although several of the noted LERs dealt with fresh fuel being placed in locations that were not appropriate per plant Technical Specifications, none of the LERs reviewed indicated any occurrence of misloading fresh fuel when the intent was to load (or move) a spent fuel assembly. Note, however, that this provides little confidence that such a misload has not occurred. It is important to recognize that the physical appearance of fresh fuel is significantly different from fuel that has been irradiated for even a relatively short period of time (less than one cycle). A fresh fuel assembly has a shiny surface, whereas the surface of an irradiated fuel assembly will have undergone slight oxidation, resulting in a dull reddish appearance. Also, visible encrustation may have built up on an irradiated fuel assembly, and the assembly may be slightly bent or twisted. However, there may be cases where plants removed a leaking or damaged fuel assembly only days or weeks after it was freshly loaded into the core and subsequently stored it on-site for months, or even years. This “slightly burned” SNF may or may not appear to be more heavily irradiated fuel. If this slightly burned assembly was mistaken for heavily burned SNF due to human or administrative error, the consequences could be serious. Procedures requiring visual inspection of the fuel before it is loaded into a cask exist at utilities to prevent the misloading of assemblies into a spent fuel cask. Frequently utility procedures require two-person independent verification of the assembly ID, and the loading procedure is often videotaped. No instances of fresh fuel being mistaken for spent fuel were found in the documentation.

Although the information reviewed for this report did not include an extensive amount of non-U.S. data, the following information was noted. In a paper included in an ANS conference proceedings,[50] the French authors noted that the Dampierre Nuclear Power Plant experienced a fuel misloading event during a core reload activity. In April 2001, a total of 113 assemblies had been placed in the wrong positions before the error was discovered. This error occurred when the 26th assembly was inadvertently substituted for the 25th assembly in the reload sequence, with the result that all of the following assemblies were also placed in incorrect locations within the core.[50] The error was detected when an operator noticed that the distribution of rod cluster assemblies in the reactor vessel was not in accordance with the expected symmetrical pattern. The cause of the misload incident was operator error. It was noted that the neutron source range monitors could not detect a local increase in reactivity unless a reactive pattern was formed close to one of the monitors. There had been no attempt to take the reactor critical because the incident was discovered before any actions were taken to reinstall the vessel head.

7 SUMMARY DISCUSSIONS

The current NRC regulatory guidance for burnup credit (ISG-8, Rev. 2) includes a recommendation for a measurement confirming assembly burnup to ensure that an incorrect loading in a burnup credit cask with a more reactive assembly than permitted by the loading criteria does not occur. A burnup measurement is recommended, in part, because of concerns regarding the completeness and accuracy of utility reactor burnup records for all SNF and concerns associated with the potential for calculation, procedural, and other human errors in selection of assemblies for loading. The purpose of this report is to detail information and issues relevant to preshipment burnup measurements when using burnup credit in PWR SNF transport and storage casks. In particular, this report reviews the role of burnup measurements in the regulatory guidance for demonstrating compliance with burnup loading criteria, burnup measurement capabilities and experience, generation and accuracy of utility burnup records, fuel movement and misloading experience, and the consequences of misloading assemblies in casks designed for burnup credit.

7.1 OUT-OF-CORE MEASUREMENTS

As it is not possible to nondestructively measure assembly fissile content or burnup directly, burnup measurement techniques/devices measure neutron and gamma radiation emitted from an assembly and use this information, along with calibration data, to infer the assembly burnup. The various techniques/devices used to measure assembly burnup were originally developed for nuclear materials safeguards purposes, have limitations in their applicability for highly accurate burnup determination, and have advantages and disadvantages relative to each other. Generally, during fork detector examinations, assemblies have been measured at only one position along their length. If the measurements at this position are representative of burnup (and the equipment calibrations against assemblies of like geometry and known burnup accurately account for axial variations in assembly burnup), the relative burnup values of the assemblies can be obtained. Previously established correlations between burnup and the passive radiation levels then verify that the measurements are consistent with operator-declared values for burnup.

Out-of-core measurements of gamma and/or neutron radiation that infer assembly burnup require calibration of the counting equipment with an assembly of known burnup and age and of identical geometry. Thus, the accuracy of the assembly burnup inferred from the gamma-ray and/or neutron measurements cannot be more accurate than the accuracy of the burnup of the reference assembly used for calibration. Additionally, the approach used in the analysis of the SNF during each measurement campaign was to accumulate measurements from a number of assemblies and generate an internal calibration by comparing each assembly with the best-derived fit of all the site data. During out-of-core examinations, any observed deviations from the best-fit curves would incorporate the uncertainties in the out-of-core measurements as well as any errors in the reactor records. The average deviations were therefore likely to be upper bounds on the random errors in the reactor records for assembly burnup. Several of the referenced reports state that the fork detector system demonstrates an average deviation from the burnup calibration curves of 2.2–5% (Refs. 4,9,14). The specific results noted in this report show the average deviation of the fork detector, when compared with calibration curves developed from reactor record burnup data, was between 2.7 and 3.5%, with maximum assembly deviations of up to 9.1%. The high-resolution gamma-ray scans were not as accurate and in the ANO campaign averaged a deviation from the calibration curve of 5.37% at the 67% confidence level. In any case, physical measurements of the assembly neutron and gamma radiation do not provide an absolute burnup measurement but instead provide a measurement of burnup based on information provided in the utility reactor records and serve to confirm the information in the reactor records. The effort to conduct the out-of-core measurement of SNF involves additional handling of the inspected assemblies, which increases the risk of assembly damage or misloading, increases the radiation dose to personnel conducting examinations, adversely impacts other fuel-handling activities, and increases the financial costs to the utilities.

Although not discussed previously in this report, subcritical measurement techniques[51] could be used for verification of subcriticality, as opposed to burnup, and hence have some relevance. However, the viability and feasibility of subcritical measurements for large systems such as transport casks have not been demonstrated and it is recognized that significant technical and operational challenges would be associated with any attempt to use subcritical measurements for verification of subcriticality in storage and transport casks.

7.2 REACTOR BURNUP RECORDS

Utility-developed assembly burnup records are based on either (1) the measured core thermal output, with burnup distributed to individual assemblies using validated computer codes, or (2) a combination of information provided by in-core detectors, measured core thermal output, and validated computer codes.

The total core power calculation has an uncertainty of about 1% (Refs. 4, 9, and 31). In the 1989 report, *The Incentives and Feasibility for Direct Measurement of Spent Nuclear Fuel Characteristics in the Federal Waste Management System*,[4] an analysis of data for Zion Unit 2, cycle 1, indicated that it is appropriate to use the extensive in-core measurements of power distribution (flux maps) to develop the end-of-cycle assembly average burnup values. Using the total core power information and in-core flux maps, assembly average burnup values were determined to within about 2% of the predicted burnup values when computer models, correctly normalized to start-of-cycle conditions and adjusted periodically on the basis of in-core measurements, were used.[4] Multiple, independent comparisons[4,5,9,12,14,34,40,41] show that the in-core measured burnups are within 1.79% and 4.2% of predicted burnups, and based on qualitative indications, the measured burnups are probably closer to the true burnups than to the nodal predicted burnups. Based on these comparisons, it may be concluded that the uncertainty in the utility-assigned burnup values is less than 5%.

One area of concern for reactor record accuracy is older fuel, including fuel that has been treated differently from the vast majority of SNF at the site. For example, some spent fuel at utilities was shipped off-site for various test programs and was subsequently returned to the site at a much later date. The records for this fuel are much more suspect than the records for fuel that has not been handled by multiple organizations. Also, as noted in Section 6of this report, Ginna and possibly other utilities store some spent fuel in consolidated rod storage canisters in the spent fuel pool. The site records may contain canister data but not data for individual assemblies contained in the canisters. If burnup is to be used as a loading criterion for such fuel, it is important that utilities verify the characteristics of the assemblies that have been handled or stored in these atypical conditions. Also, "batch-average" assembly burnup data, which is much less accurate than individual assembly burnup data, has been used in at least one case and should not be used in operations to support the loading of assemblies into burnup credit transport casks where loading criteria include assembly-specific burnup values. The core-follow and in-core monitoring data for SNF is stored by utilities and should be used to reconstruct assembly-specific burnup data where appropriate.

7.3 MISLOADS IN A PWR BURNUP CREDIT CASK

The analyses in the report *Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask*[45] show that fresh fuel misloads into a transport cask can have a significant effect on criticality safety. For misload conditions involving fresh fuel assemblies, the results indicate that misloading a single fresh assembly with 3, 4, or 5 wt % ^{235}U enrichment results in an increase in k_{eff} of approximately 2, 4, or 6%, respectively. Notably, a single fresh fuel assembly with 5.0 wt % ^{235}U enrichment loaded into the cask center will result in an increase in k_{eff} of more than 0.05, while misloading two fresh 5 wt % ^{235}U assemblies results in an increase in k_{eff} of more than 0.10. Also, misloading two assemblies that are underburned by 75% results in an increase in k_{eff} of 0.025–0.035, while misloading four assemblies that are underburned by 50% results in an increase in k_{eff} of 0.025–0.035.[45]

Concerning the possibility of mistaking a fresh assembly for an irradiated assembly when loading a transport cask, it is important to note that the physical appearance of fresh fuel is significantly different from fuel that has been irradiated, even for a relatively short time (less than one cycle). A fresh fuel assembly has a shiny surface, whereas the surface of an irradiated fuel assembly will have undergone slight oxidation resulting in a dull reddish layer. Also, visible encrustation may have built up on an irradiated fuel assembly, and the assembly may be slightly bent or twisted. "Slightly burned" fuel that has received only days or weeks of irradiation may exist in some utilities' spent fuel pools. This slightly burned SNF may or may not give the appearance of more heavily irradiated fuel. If this slightly burned fuel were mistaken for heavily burned SNF due to human or administrative error, the consequences could be a serious impact to the safety margin.

Rigorous and comprehensive procedural requirements for preparing, reviewing, and conducting fuel transfers to ensure that the proper fuel assemblies are loaded into the correct cask position could suffice to preclude a fuel misload that violates the criticality loading curve criteria. For example, two fuel-handling operators should independently verify assembly identifiers at each stage of the burnup verification and cask loading procedures. Reactor record assembly burnup information should be independently verified. This can be done by independent out-of-core measurement of the assembly or by verifying the original core-follow burnup data against the in-core measurement data. These procedural measures ensure proper assembly selection and records assignment.

Procedures requiring visual inspections and/or simple field measurements (e.g., gross radiation measurements, Cerenkov radiation detection) to be performed before loading SNF into a cask may be sufficient to prevent fresh or nearly fresh fuel from being mistakenly loaded into a burnup credit cask. Any additional measurement program would have to be evaluated to determine its limitations to detect the various ages and irradiation levels of SNF. For example, the DCVDs may not be able to differentiate between very old, moderately burned SNF and new, very lightly burned SNF. Also, equipment alignment is critical to Cerenkov examinations and is especially critical when examining fuel that has been cooled for long periods of time. Near-neighbor assemblies and pool water quality (boron concentration) can also affect the Cerenkov examination results.

7.4 MISLOADING EVENTS

A review of known fuel misloading experience indicated that utility fuel-handling errors are rare but have occurred. More than 1.5 million fuel-handling activities have occurred since the NRC LER tracking system was developed, with 19 LERs noting assembly misload events. These included 18 LERs for assemblies not meeting the Technical Specifications for the locations in which they were stored and 1 LER involving moving an incorrectly identified assembly. Two misloading events involving SNF casks were identified. It should be noted that the database of LERs covers a limited period of spent fuel pool movement experience, and that it is not clear that all misload events that occurred during that period are accurately reflected in the database. Given that this is the case, it is difficult to draw quantitative conclusions about spent fuel pool misload frequency and trends. Considering the number of fuel-handling activities, the frequency of reported misload events is low.

7.5 ISSUES RELATED TO OLDER SPENT NUCLEAR FUEL

Utilities use various NRC-approved methods to determine the end-of-cycle assembly average burnup values that are recorded in reactor records. However, the methodologies used have changed over time and older data are potentially less accurate than newer data. Utility use of flux map data, design codes (and code constants), and full-core measured/inferred radial power distributions has improved over time, decreasing the burnup uncertainty of newer spent fuels. Because burnup values of older fuels were not recalculated with these incremental improvements, the burnup uncertainty factors used for older fuels may be larger than those for newer fuels by an undetermined amount. Other characteristics of older SNF include less complex fuel designs, lower initial assembly enrichments of ^{235}U (2–3%), and lower assembly average burnup values. Because utilities have not uniformly or consistently calculated SNF burnup using the same methods, it may be useful for the industry to provide information on the codes and methods that have been historically used to develop assembly reactor record burnup information.

Also, the extent and/or rigor of QA verification given to the data entry, transcription, and information verification may be inconsistent among utilities. A number of utilities have already made commitments to NRC to develop and implement plans to validate databases used in the control of spent fuel pool storage. As the utilities migrate their SNF databases to newer software, they frequently conduct verification programs to ensure that the data are transcribed correctly. As noted in several of the referenced LERs, these verification programs can involve confirmation of the utility spent fuel pool configurations by verifying assembly IDs and assembly burnup against Technical Specifications for assembly location within the spent fuel pool. This verification of both the spent fuel pool configurations and the SNF databases serves to validate not only the assembly locations within the spent fuel pool but also the accurate assignment of the burnup and cooling time to the proper assemblies in the reactor records.

7.6 METHODS FOR CALCULATING BURNUP

A number of existing reports discuss varying methods and criteria for developing a “final” assembly burnup value when there is a difference between reactor record burnup values and out-of-core burnup measurement values.[7,14,32,52] Two methods for using reactor records of assembly burnup for developing this final assembly burnup value are as follows.

- Method 1—Use the burnup value from the reactor record, when confirmed to be within a predefined range by an out-of-core burnup measurement, reduced by the uncertainty in the reactor record.

- Method 2—Use the burnup value from an out-of-core burnup measurement, as calibrated by the reactor record data, reduced by the uncertainty in the out-of-core burnup measurement, as determined statistically from all of the measurements in the calibration.

Each of the two methods relies on a calibration of the measurement system by measuring a large number of geometrically identical assemblies and using reactor record burnup data for these same assemblies to establish the calibration.

The first method has the advantage of simplicity in that the assembly burnup can be determined before out-of-core measurement from reactor data and is subject only to confirmation by the out-of-core measurement. The second method has the advantage that both the burnup and its uncertainty are measurement based.[52]

A third possible method using a less robust measurement system than that required in method 2 above would involve using verified reactor burnup records, which would be required to have a clearly defined and quantified uncertainty optionally coupled with a simple measurement (e.g., Cerenkov light measurement) to prevent the misloading of fresh and slightly burned fuel assemblies. This method would have the advantages of simplicity, potentially no additional assembly-handling operations, and the confirmation that fresh fuel is not mistaken for spent fuel.

8 OBSERVATIONS

Over the past 20 years, multiple SNF out-of-core examinations have been conducted at U.S. utilities to compare utility records for assembly burnup and cooling time with out-of-core measurement data. The measurement equipment used [neutron/gamma-ray detection systems (e.g., the "Fork Detector" system from Los Alamos National Laboratory) and HRGS systems available commercially from several vendors] was developed to detect fissile material diversion and was not specifically designed for ease of use in reactor spent fuel pools or optimized to accurately verify or quantify SNF burnup independent of utility reactor records. However, the data gathered from these examination campaigns have been fairly uniform and consistent and are useful for evaluating the effectiveness of these measurement techniques/equipment and the general accuracy of reactor burnup records. Although the out-of-core burnup measurement techniques and equipment used in the examination campaigns are rather dated, published information on SNF burnup measurement campaigns involving more modern techniques/equipment could not be found.

In addition to the out-of-core measurement campaigns to evaluate assembly burnup versus reactor records, some utilities have retroactively applied significant effort to "verify" the data in their SNF records. They have done this by comparing the assembly burnup records with burnup calculations performed using NRC-accepted design codes and in-core monitoring measurements including consideration of cycle-specific total power output and full-core measured/inferred radial power distributions. In a number of cases, utilities have upgraded the software used to track their SNF and have used this opportunity to verify the data input accuracy of their reactor records. Other utilities have reverified their assembly records to ensure that the burnup information on their spent fuel pool assemblies matches the burnup calculations calculated for the fuel when it was removed from the reactor cores.

The following observations highlight key points from the information reviewed for this report.

<u>Reactor Burnup Records</u>

- Utility records for fuel burnup are based on either (1) the measured core thermal output, with burnup distributed to individual assemblies using validated computer codes, or (2) a combination of information provided by in-core detectors, measured core thermal output, and validated computer codes.

- There is a significant amount of data available from 1980 to the present to support a finding that utility records for fuel burnup are accurate for individual spent fuel assemblies to at least 5% of "true" assembly burnup. These data originate primarily from reactor core-follow data, ex-core burnup measurement programs, comparisons between calculated burnup values (on which reactor record values are based) and burnup values inferred from in-core measurements, and retrospective evaluations based on comparisons between existing reactor record values and calculated values.

- Utilities do not all use the same methods to calculate and verify the assembly burnup values that are recorded in their reactor records. Moreover, the computational methods used by many utilities have evolved over time such that burnup values for older and newer fuel are based on different methods. Hence, for SNF assemblies to be transported in burnup credit casks, utilities need to demonstrate how the burnup values in the reactor records were developed and recorded and document information as to the accuracy of their recorded burnup values. Examples of this type of activity are the reactor record verification programs of Duke Energy and TVA described in this report.

- In at least one case, burnup values in a utility's reactor records for some assemblies are based on the "batch average" burnup, which is definitely not an accurate representation of the individual burnup for each of the assemblies. The number of utilities that may have used batch average data is unknown. However, based on a review of various data sources for discharged fuel, it is expected that the use of batch averaged values is limited in number and would apply only to older fuel records. Contemporary reactor records use individual, assembly-specific burnup calculations and values.

- Unless a QA program is implemented to ensure the accuracy of reactor records, reactor records may become less accurate over time as a result of transcription errors, as record media degrade or are changed to newer media, or as software is updated.

Burnup Verification Measurements

- Out-of-core measurement systems can adequately verify reactor record burnup information. However, as is also the case for in-core measurement systems, out-of-core measurement systems cannot measure fuel burnup directly. Instead, these out-of-core measurement systems measure gamma-ray and/or neutron emissions from the assemblies, which are then compared to a calibration curve to develop an estimated fissile content and corresponding assembly burnup.

- The burnup measurement programs reviewed for this report concluded that out-of-core measurement systems provided somewhat less accurate burnup values, as compared to reactor record burnup values developed using reactor in-core monitoring systems and design codes. The modern in-core systems, which use core measurements and some data from design codes, produce burnup values that are generally within approximately 2% of the burnup values predicted by design codes, whereas the out-of-core measurement systems produced average assembly burnup results that were within the expected fork detector accuracy range of 2.2–5% of predicted burnup. However, there were some fork detector examinations with maximum assembly deviations as high as 9.1%. Regardless, these examination programs demonstrated that out-of-core measurements could identify substantially underburned fuel assemblies and hence could be used to prevent them from being accidentally loaded into spent fuel transport casks.

- Careful calibration against an assembly of known burnup, known cooling time, and identical geometry is required to achieve the reported accuracies with the fork detector used in the U.S. studies. Thus, in practice, these out-of-core measurements are dependent on and calibrated against reactor burnup records. Consequently, the burnup values inferred from these measurements tend to have a higher uncertainty than reactor record assembly burnup values determined using in-core measurements.

- Fuel assembly axial-burnup profiles have a significant impact on reactivity and are therefore an important component in determining "average assembly burnup." For the fork detector examination programs evaluated in this report, personnel used measurements at the assembly centerline (midplane) and assumed that the axial profile of the reference assembly could be used to estimate the assembly average burnup (i.e., it was assumed that the axial profiles were the same for the reference and measured assemblies). This approach could give erroneous burnup measurement results for assemblies that have different axial burnup characteristics. Hence, if out-of-core measurements are used, care should be taken to ensure that correct average burnup information is collected, commensurate with the measurement accuracy goals and criteria for approved contents.

- The costs and risks associated with performing out-of-core burnup measurements should be balanced with the risks and potential costs of not performing the measurements. Out-of-core measurement campaigns require utility resources for planning and execution, increase the dose to personnel, increase the risk of damage to assemblies and potential fuel mishandling events due to the increased assembly movements, and have associated financial cost to the utility. The risks and potential costs associated with loading a significantly underburned assembly into a transport or storage configuration have not been explored in this report.

- The neutron-counting measurement systems appear to be more accurate than the HRGS systems and require less skilled operators for handling the SNF assemblies.

- Considerable efforts, primarily motivated by interests related to nuclear material safeguards, are ongoing to develop better and more accurate measurement systems.

Consequences of Fuel Assembly Misloading

- The consequences to k_{eff} of loading assemblies that have slightly reduced burnup (e.g., 5% due to uncertainties in the burnup verification process), as compared with the required burnup, are fairly small ($\leq$ 1%). On the other hand, loading one or more highly enriched (i.e., > 4 wt %) fresh fuel assemblies has a significant consequence on criticality safety. These findings suggest that while it may not be overly important to precisely verify the burnup value, it is important to ensure that fresh or very-low-burnup (i.e., nearly fresh) fuel assemblies are not misloaded into a cask.

Fuel Movement and Operational Considerations

- Although utilities' record of reliability in selecting and moving assemblies during fuel-handling processes may be characterized as "good" from the standpoint that no inadvertent criticalities have occurred as a result of fuel misloading, there are a number of documented examples of fuel misloading. Considering the large number of fuel assembly movements that have been executed, relatively few mishandling events involving movement of an incorrectly identified fuel assembly have been reported. Procedural violations related to SNF movements that resulted in violations of plant Technical Specifications have occurred with a fairly low frequency. Most of these events were not a result of incorrect burnup values assigned to the SNF but were instead the result of personnel error in selecting assemblies for movement.

- Fresh fuel is visibly different from SNF due to the oxidation, crud buildup, and bending/twisting of the latter, so visual inspection should easily differentiate new assemblies from SNF. However, there is some uncertainty about the appearance of an assembly with very limited burnup (e.g., removed promptly after start-up due to leaking fuel rods or some other problem) after it has resided in a spent fuel pool for a number of years.

- Visual inspections and/or simple field measurements (e.g., gross radiation measurements, Cerenkov radiation detector) could be performed during cask loading to detect and prevent accidental loading of fresh or nearly fresh fuel. A basic detector system could be devised to ensure that an assembly has some minimum activity level. One examination tool that may suffice is the DCVD. Any measurement program would have to be evaluated to determine its limitations in detecting SNF. For example, the DCVDs may not be able to differentiate between very old, moderately burned SNF and very-lightly-burned (i.e., nearly fresh) SNF. Also, any examination limitations, such as equipment alignment for Cerenkov examinations, must be evaluated to ensure that adequate examination results are achievable for all SNF to be evaluated.

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OCTOBER 2009