Nuclear Fuel Burnup Plant Records: Generation and Accuracy

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Abstract: An accurate evaluation of the spent fuel assemblies burnup is essential to the allowance and implementation of a burnup credit program. One of the recommendations in the U.S. Nuclear Regulatory Commission (NRC) Interim Staff Guidance 8 (ISG-8) Revision 2 is an out-of-core measurement to confirm the reactor record and compliance with the assembly burnup value used for cask loading acceptance. The NRC is currently evaluating potential alternatives to the out-of-core measurement recommendation in ISG-8. This evaluation includes the determination of the reliability and accuracy of the burnup values from the reactor records and whether these values are acceptable for use in burnup credit and what additional requirements and safeguards would be needed to allow their utilization. The fuel assembly burnup is a core-follow parameter that is treated and controlled as safety-related data and is used extensively in reactor design and in safety analysis. It is also used in regulation compliance in the on-site spent fuel management. This paper describes the instrumentation and the measurement processes, as well as the computer codes that are used to generate the reactor records in a typical pressurized water reactor (PWR) in the United States. It also reviews the parameters important to the accuracy of these records and the variability in flux map measurements resulting from the successive methodology and code changes over the past two decades.

1. Introduction

The fuel assembly burnup is a core-follow parameter that is used by plant personnel in core design activities as well as regulation and procedure compliance. The nuclear utilities are required by existing NRC regulation to ensure that the irradiated fuel remains below an assembly average burnup limit of about 45,000 MWD/MTU to protect the fuel integrity. 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. In current industry practice, a very important use of the burnup values is in 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 fuel assembly burnup is simply the amount of cumulative energy generated per weight of fuel. Like many important core reactor physics parameters, the burnup is a defined quantity that cannot be directly measured but is inferred from a combination of plant measurements and reactor physics codes calculations. The process that yields this inferred fuel assembly burnup value of record is illustrated in Figure 1.

The burnup cumulated in fuel assembly i, $\Delta B_{i,}$ during a reactor operating time interval Δt can be expressed as:

$$\boldsymbol{\Delta B_i = [(RTP * \Delta t * CF) / M_F] * P_i}\ , \qquad (1)$$

where ΔB_i is in megawatt–days per metric ton of uranium (MWD/MTU) and Δt is in days. RTP is the reactor rated thermal power or “full power”, a constant for a given core, expressed in megawatts (MW), 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.

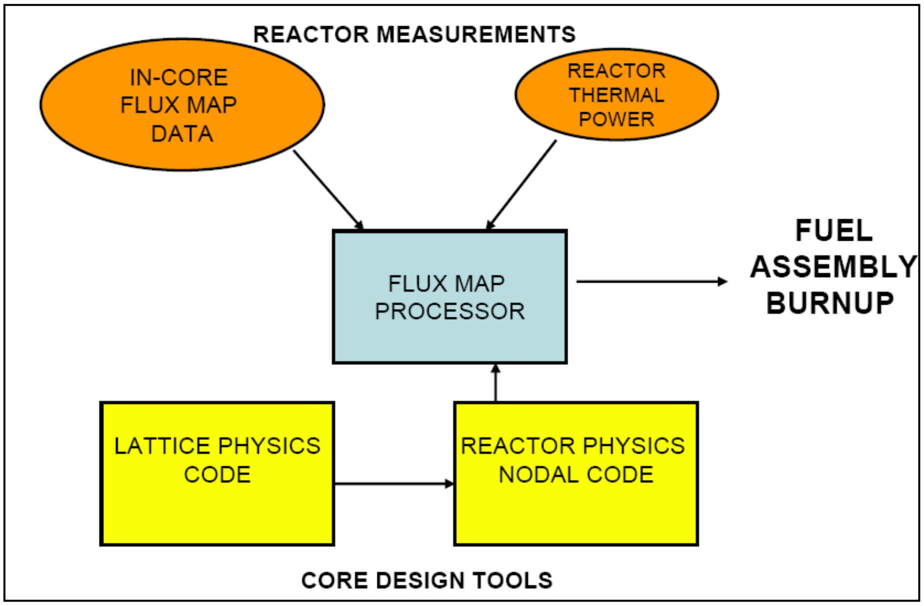

Fig. 1. Assembly Burnup Determination Flow

In Eq. (1), 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. The measured reactor thermal power is determined from temperature and flow measurements of the reactor coolant circulating through the reactor core. The primary components of the reactor 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). Standards from the American National Standards Institute (ANSI) and utility operating procedures 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.

All parameters in Eq. (1) are measured quantities except P_i which is inferred. The uncertainties associated with the measured quantities are quite low; therefore, an accurate determination of the fuel assembly burnup requires accurate determination of the core power distribution.

2. Determination of the Core Power Distribution

The power distribution is determined by NRC-approved reactor analysis codes and is regularly verified using flux map measurements. The inferred core power distribution is obtained from a combination of in-core measurements taken during a flux map and corrected predictions from a nodal simulator code. Since approximately 30% of the core is instrumented, assembly-to-assembly coupling factors are used to infer the reaction rates and power at the non-instrumented locations based on the flux measurements in the nearby instrumented assemblies. These coupling factors are generated from cycle-specific, burnup-dependent computer calculations. Due to the permanence and the random 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 three to four reactor cycles.

Due to 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. A typical PWR movable in-core instrumentation configuration is illustrated in Fig 2. Other reactor types have fairly similar systems. 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. 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.

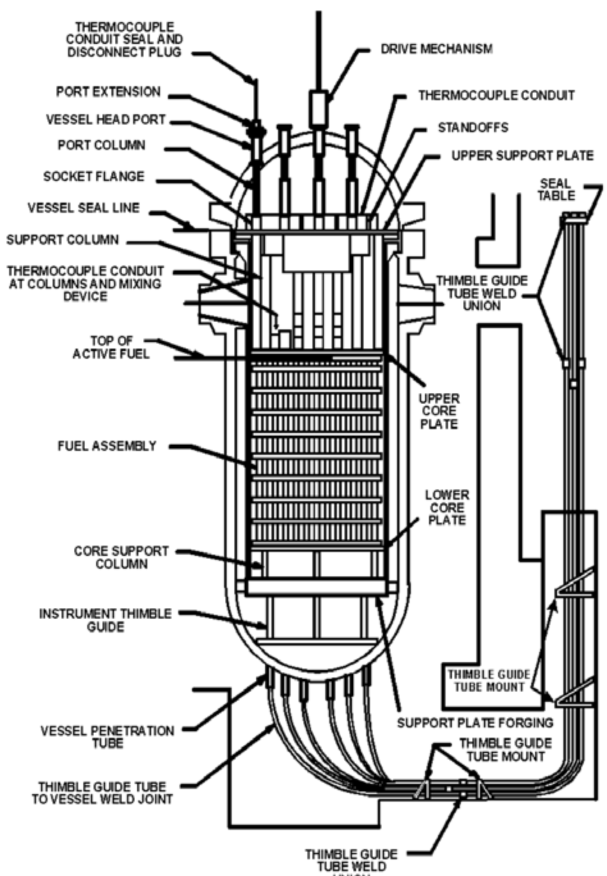

Fig. 2. Typical Westinghouse pressurized water reactor in-core flux map system.

3. Flux Map Processing

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 non-instrumented locations.[1,2,3] These 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 non-instrumented locations to the nearest instrumented assemblies. 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 non-instrumented assemblies.
- Coupling or weighing coefficients for each non-instrumented location are derived using data from the nearest instrumented assemblies.
- The power distributions at the non-instrumented 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.

Eq. (1) is then used to determine the cumulative burnup from the inferred core power distribution and the cumulative reactor energy generation.

Figure 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 at the non-instrumented locations to yield accurate power distributions, and thus accurate thermal limits margins.

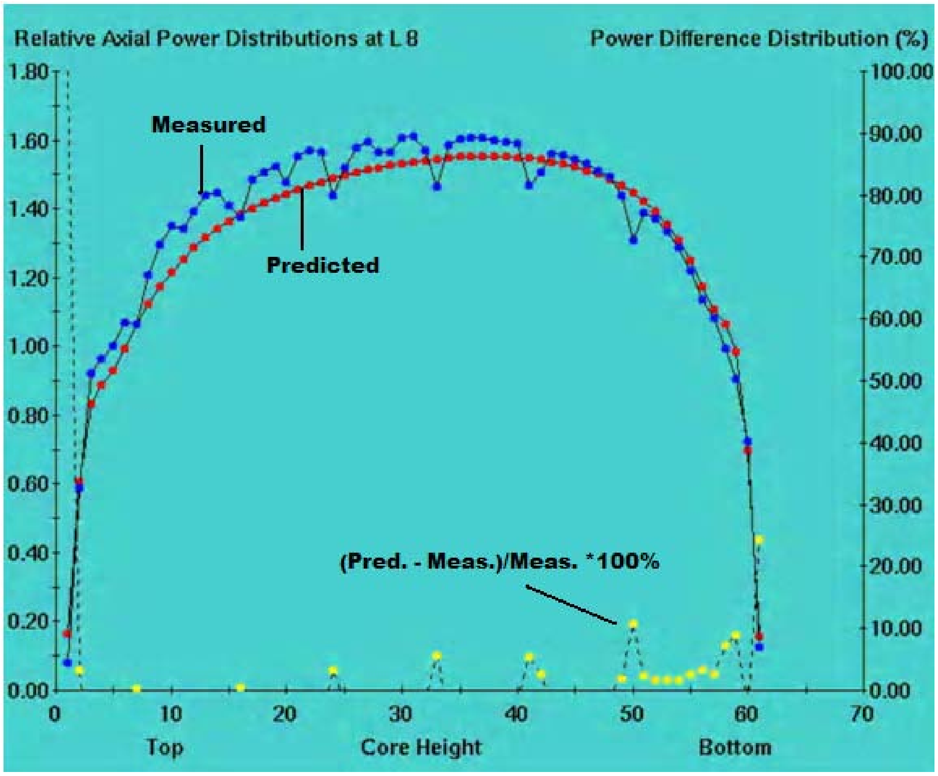

Fig. 3. Actual flux map data illustrating the typical comparison of measured vs. predicted axial power distribution at an instrumented core location. (Courtesy Progress Energy)

4. Validation of Core Physics Codes

The current methodology used to generate the core-follow burnup 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[4]) for fuel assembly modeling and a nodal reactor simulator code (such as SIMULATE[5]). 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. 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 depletion step. 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.

The primary motivation for accuracy in fuel assembly burnup measurement stems from the regulatory-based requirement for accurate core reactivity prediction at all power conditions during the operating cycle. 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. 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 shutdown margin (SDM), a calculated parameter that varies with cycle exposure, is accurately and conservatively determined. 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. 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. 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.

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 operation 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 an acceptable level of confidence in the fuel assembly burnup values used in the reactor reload design process.

5. Burnup Measurement Accuracy Evaluation

There is an extensive database of pairs of measurements and predictions at the core instrumented locations as they are generated on a monthly basis during flux mapping. Comparisons of measurements to predictions are performed to determine the nodal simulator code accuracy in calculating the power distribution and, thus, the burnup of the individual instrumented assembly. The flux mapping process generates a summary report which contains a detailed statistical evaluation of the measurements, and predictions at the instrumented assemblies, and of the inferred data at the non-instrumented locations. As part of the license approval process, the vendors of the flux map processors provide NRC with proprietary licensing topical reports documenting the uncertainty associated with the 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.

A study conducted for the Department of Energy (DOE) Yucca Mountain Project [6] evaluated the reactor records of 5447 PWR fuel assemblies from nine PWR plants. Comparison of core design values and flux map measurements indicated an uncertainty in burnup of between 2 to 4.2%. Another study, performed by the Electric Power Research Institute (EPRI)[7], evaluated the core-follow code SIMULATE performance in predicting the assembly average burnup values by accessing data from actual flux maps from three consecutive cycles and using the measured reaction rates at the instrumented locations only. 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 study also looked at detector reproducibility to evaluate the statistical fluctuations associated with repeated measurements of a signal detector in a given location. The mean error developed from this analysis was –0.147% with a standard deviation of 0.748%.

Nuclear utilities have periodically performed burnup evaluation studies. Duke Energy Corporation conducted a statistical evaluation[8] of its fuel assembly burnup database to quantify the bounding burnup measurement uncertainty. The study used the data from the entire discharged spent fuel inventory (about 1,900 assemblies) from its McGuire plant’s two reactors. The maximum observed individual assembly error between the core follow code predictions and in-core flux map measurements was about 4.0%. The study noted a clear measurement uncertainty decrease with higher discharge burnup.

Another large scale utility evaluation was that of the Tennessee Valley Authority (TVA) which wanted to validate its spent fuel database consisting of 1,117 fuel assemblies by identifying any errors due to transcription, data entry, or others factors. Comparison of the flux map generated burnup measurements and the core design code burnup predictions were conducted. For assemblies with burnup value differences greater than 2% a new burnup calculation was performed and the database entry was revised. 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. Most burnup errors were less than 1 GWd/MTU, and the largest observed error was 1.51 GWd/MTU on an assembly with a burnup of approximately 41.8 GWd/MTU, or 3.6% difference. Only seven assemblies failed the 2% acceptance criterion and had to be revised.

Record keeping requirements at individual utilities have evolved over the past few decades, as have the pertinent regulatory requirements by NRC. For old spent nuclear fuel dating to the 1960's and 1970's, the reactor records at some plants may include "batch averages" burnup values instead of individual assembly burnup values. A study [9] for one such plant involved 12 discharged fuel reloads totaling approximately 740 fuel assemblies and yielded mixed results. About one third of the burnup values were reported as "batch average" and had differences between in the + 5 to -5 GWd/MTU range, with some assemblies exhibiting differences as high as -16 GWd/MTU. It is expected that utilities have sufficient information from their archived flux map records to determine the individual burnup values for the assemblies that currently have batch average values.

One of the main goals of the nuclear utilities core follow activities is to assess the accuracy and adequacy of their core analysis codes. The accuracy of total core burnup measurements is easily evaluated by monitoring the change in boron concentrations in the core throughout a fuel cycle and comparing it to the code predictions. There is very good agreement throughout the industry between the measured and the predicted boron concentration values as illustrated in Fig. 4. For a typical PWR with a beginning of cycle boron concentration of about 1600 ppm and boron depletion of 3.5 ppm/day, the impact of a miscalculation of the total core burnup by 5%, would equate roughly to missing the end-of-cycle full-power capability by more than 22 days, which would be very obvious. By monitoring the boron concentration in the core, a utility can show that integral core power, and thus true core burnup, has been calculated accurately.

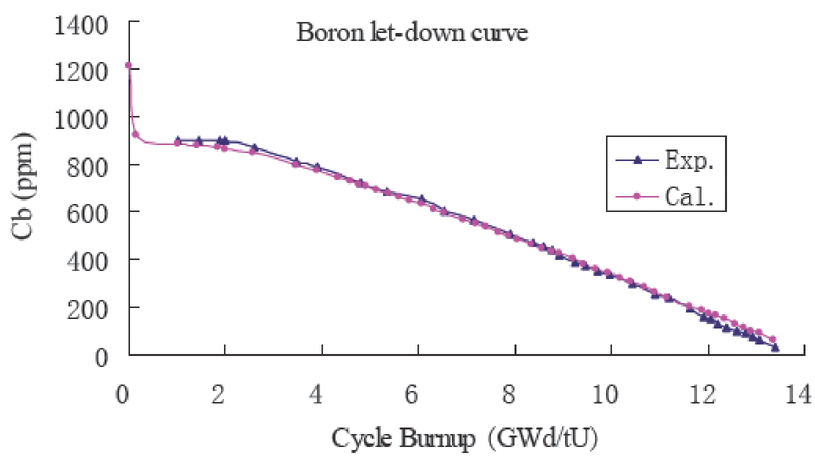

Fig. 4 Typical boron concentration variation with cycle burnup

Another factor affecting the accuracy of fuel burnup calculations is the increase in reactor residence time (increased fuel burnup) which has an equalizing effect on burnup uncertainty. An analysis[7] conducted on discharged fuel from three cycles of operation shows a general decrease in uncertainty with an increase in reactor residence time. The mean burnup uncertainty was 1.9, 0.98, and 1.02% for one, two, and three cycles of burnup, respectively. Another factor affecting the burnup calculations is the use by the industry of more precise feedwater flowmeters, which reduces the uncertainty in determining feedwater flow, and thus the reactor thermal power. The calculated thermal power is an important contributor to the accuracy of the individual assemblies burnup values as indicated in Eq. (1).

The evolution of the reactor analysis computer codes used to calculate the core power distribution had a significant impact on the uncertainties in assembly burnup calculations. The successive improvements in these codes allowed for more accurate core behavior predictions and for continuous real-time monitoring capability.

6. Conclusion

Accurate determination of the assembly burnup depends on an accurate determination of core power distribution. The core power distribution is adequately predicted by validated and approved core design codes and in-core reactor measurements are used on a regular basis to update this prediction to reflect the actual reactor operating conditions. Commercial flux map processing codes demonstrated in their licensing submittals that the resulting power distributions were highly accurate, and utility reactor records appear to be accurate to well within 5%. Multiple independent comparisons performed at different utilities in the past couple of decades involving several thousand in-core measured assembly burnup values report reactor record deviations of less than 4% from core design predictions. One exception involves old fuel records dating several decades from some utilities which reported batch average burnup values instead of individual assemblies burnup data.

A notable challenge to burnup record accuracy stems from the fact that industry records do not take into account any errors in recording or transcribing the assembly burnup data into utility databases. An additional challenge is that of the successive migrations from older or obsolete record systems into newer databases. Spent fuel inventory assessments need to be performed to identify any record discrepancies.

REFERENCES

1. C. L. Beard and T. Morita, *BEACON Core Monitoring and Operation Support System*, Westinghouse Electric Corporation, WCAP-12472, August 1994.
2. G.R. Correll and C. A. VanderMass, *Power Distribution Uncertainty for INPAX-W in Westinghouse Plants,* Siemens Power Corporation, EMF-93-164, September 1993.
3. A. J. Harris and K. A. Jones, *The INCORE-3 Program,* Westinghouse Electric Company, WCAP-8492, 1975.
4. M. Edenius, *CASMO-3 A Fuel Assembly Burnup Program User's Manual,* NFA-89/3, Studsvik NFA-89/3, 1991.
5. J. Cronin et al., *SIMULATE-3 Methodology Advanced Three-Dimensional Two-Group Reactor Analysis Code,* STUDSVIK/SOA-95/18 (1995).
6. H. L. Massie, Jr., *Reactor Record Uncertainty Determination,* AREVA/FANP Document No. 32-5041666-02, Framatome ANP, Inc., May 26, 2004, DOC. 20040623.0002.

7. Electric Power Research Institute, *Determination of the Accuracy of Utility Spent Fuel Burnup Records – Final Report*, EPRI TR-112054, July 1999.

8. "Duke Energy Corporation (Duke), McGuire Nuclear Station Units 1 and 2, Docket Numbers 50-369 and 50-370, Technical Specifications Amendment, Request for Additional Information (RAI); TS 3.7.15–Spent Fuel Assembly Storage, and TS 4.3–Fuel Storage," correspondence from G. R. Peterson to NRC Document Control Desk, dated April 22, 2004.

9. Bechtel SAIC Company, LLC, *Source of Burnup Values for Commercial Spent Nuclear Fuel Assemblies, Rev. 3*, December 2004.