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AEC Research and Development Report

STATUS OF BURNUP
CALCULATIONAL TECHNOLOGY AS
APPLIED TO THE NUCLEAR MATERIALS
SAFEGUARDS ACCOUNTING SYSTEM

June 30, 1971

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STATUS OF BURNUP CALCULATIONAL TECHNOLOGY AS APPLIED TO
THE NUCLEAR MATERIALS SAFEGUARDS ACCOUNTING SYSTEM

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I. INTRODUCTION

Burnup calculational technology is an important tool for nuclear materials safeguards systems. In a previous document⁽¹⁾, the authors surveyed the function of the nuclear reactor in the nuclear materials safeguards system. This document is an extension of that work and is specifically concerned with the degree to which burnup technology can be improved so that it can be more usefully applied to nuclear materials safeguards problems.

In assessing how accurately power reactor core isotopic inventory can be predicted, one needs to consider several factors. These include:

- Operational codes and standards applied to the power plant
- Reactor instrumentation
- Calculational limitations
- Theory-experiment correlations.

These factors are discussed in detailed in this report.

The above factors are used to develop, in tabular form, future capability and current performance standards for the burnup problem. These tables, summarized in the next section, constitute a statement of both what the status of inventory calculational capability is at the present time and what one can reasonably expect in the future.

Data from reactor operation relevant to the nuclear materials safeguards program are the measures of thermal reactor power versus time, and neutron flux levels in the reactor core. In addition one has the outputs of applicable computer programs which utilize the above data to follow fuel exposure and predict depletion and buildup of heavy metals. To use this

data correctly, accurate information on the isotopic distribution and total isotopic content of each fuel unit must be supplied to the utility by the fuel vendor.

The utilities' increasing responsibility attendant with nuclear fuel ownership, and the substantial investment in their nuclear fuel, has warranted correspondingly extensive efforts in core analysis by the utilities. This is particularly true of those utilities with several nuclear reactors. Consequently, the newer power stations have included in their reactor designs increased capability to measure the critical parameters, namely thermal power and core neutron flux, and have increased their staffs for in-house interpretation of data and construction of models and software programs that can be tailored to reactor behavior to predict depletion and buildup of the heavy metal isotopes.

The availability of better data from the nuclear power industry, along with the increasing competence of the utility staff personnel in interpreting this data, particularly in the area of isotopic inventory calculations, is a most hopeful sign. This phenomenon, when coupled with the current and anticipated state of the technology in burnup calculations, indicates that significant contributions can be expected from burnup technology as applied to the nuclear materials safeguard program.

II. SUMMARY

Two operational standards are being developed which will result in the development of acceptable operational data for use in studying nuclear fuel burnup. By extension, this data will make the nuclear safeguards role much easier to perform.

One such effort is being brought about by an ASME committee and will be codified as PTC 32.1 for nuclear steam supplies and PTC 32.2 for nuclear reactor fuel. The other effort is the guidelines developed by the USAEC as published in Appendix B to 10 CFR 50 and titled "Quality Assurance for Nuclear Power Plants".

The ASME code recommendations are summarized in Table I.

TABLE I

PTC 32.2 NUCLEAR FUEL INVENTORY REQUIREMENTS

<u>Item</u>	<u>Allowable Error</u>
Initial Mass of Uranium Fuel	$\pm 0.25\%$ (95% confidence)
Initial Mass of Plutonium Fuel	$\pm 1.0\%$ (95% confidence)
Discharged Mass of Uranium Fuel	$\pm 0.5\%$ (95% confidence)

These codes, when they become engineering practice, are going to result in standardization of limits on fissile inventory which are low enough to make reasonable contributions to the nuclear safeguards problem.

Power reactor instrumentation accuracy is important to the nuclear materials safeguards role. In general, reactor process instrumentation has error bounds in the neighborhood of $\pm 4\%$. Nuclear instrumentation

errors tend to be somewhat larger ($\pm 5-10\%$). The errors in measured nuclear parameters, made up in part by instrument errors, also tend to be in the range of ($\pm 5-10\%$). These accuracies have an important bearing on the nuclear safeguards role. In particular, any nuclear safeguards system which includes power reactor measurements must recognize the current limitations associated with this instrumentation.

There are certain inherent limitations in the currently available calculational models which might be applied to the nuclear materials safeguards system. In general, isotopic inventory can be pre-calculated to $\pm 5\%$ at the 95% confidence level. In this type of calculation, the largest error on any fissile or fertile isotope in the core is of the order of $\pm 8\%$ at the 95% confidence level.

Theory-experiment correlations are used to pinpoint areas of needed development in theoretical calculational methods. The current state of the art can be summarized assuming reactor power is known to $\pm 10\%$. If this is the case, then, U^{235} is known to $\pm 3\%$, total Pu to $\pm 6\%$, U^{236} to $\pm 6\%$, and U^{238} to $\pm 0.1\%$. These numbers represent a reasonably accurate statement of several different detailed observations which have been made on several different reactor cores. They are summarized in Table II along with the other pertinent fissile and fertile isotopes.

TABLE IICURRENT AND PROBABLE FUTURE UNCERTAINTIES IN PREDICTED
ISOTOPIC CONTENT OF POWER REACTOR CORES

<u>Parameter</u>	<u>Probable Future Capability**</u>	<u>Current Performance*</u>
U ²³⁵	1.5%	3%
U ²³⁶	3%	6%
U ²³⁸	0.1%	0.1%
Total U	0.1%	0.1%
Pu ²³⁹	3%	5%
Pu ²⁴⁰	6%	8%
Pu ²⁴¹	9%	11%
Pu ²⁴²	11%	15%
Total Pu	3%	6%

* Assuming $\pm 10\%$ error in thermal power

** Assuming factor of two increase in accuracy of thermal power measurement.

III. DISCUSSION

A. Codes and Standards

Reactor operators are improving and developing standards for use throughout the industry which will improve nuclear fuels accounting. One such effort is represented by an ASME Standards Committee with representation from utilities, fuel suppliers, and reactor suppliers. This committee was organized in 1965 with the objective to develop two testing codes; PTC 32.1 for Nuclear Steam Supply Systems and PTC 32.2 for Nuclear Reactor Fuel.^(2,3) The former, which has been approved by ASME, has as its goal to establish procedures for conducting tests to determine the thermal performance of a nuclear steam supply system. The latter, PTC 32.2, on which the committee is now working, has as its goal to specify those measurements that may be required to evaluate the energy generation performance of nuclear reactor fuel, and the associated transmutation and depletion of heavy elements.⁽⁴⁾ This is generally the type of effort required in order for the nuclear power plant to play a role in the safeguards nuclear material accounting system. Several other joint efforts are underway within the industry, supported by AIF and EEI for example, and are contributing to increased capability in nuclear fuels accountability.^(5,6)

In addition, the USAEC has provided guidelines to the utilities, one of the more recent of which has been Appendix B to 10 CFR 50 - Quality Assurance Criteria for Nuclear Power Plants, which charges the operator with rather extensive quality assurance responsibilities. One of the consequences of these added responsibilities is that the utilities expect to

have more information on fuel performance, including isotopic content, and to have more involvement with the reactor suppliers on the design of nuclear power plants. This involvement should include the development of equipment for determining flux and power. Such a program could result in increased capability for the utility in their accounting for nuclear fuels.

The ASME code recommends that the initial total mass of uranium and plutonium of a unit of fuel (one or more fuel assemblies) prior to irradiation shall be determined individually for each fuel assembly from the weight ratio of uranium and plutonium to the total mass of fuel material in the assembly, measured and recorded during the process of manufacture, and that the methods used to determine the mass of uranium in the fuel shall have an uncertainty no greater than $\pm 0.25\%$ at the 95% confidence level for a batch of fuel consisting of more than 3000 kilograms of uranium. In addition, for plutonium fuel, the methods shall produce an uncertainty no greater than $\pm 1.0\%$ at the 95% confidence level for a batch of fuel consisting of more than 500 kg of mixed uranium plus plutonium. It appears that fuel vendors are willing to certify routinely, uranium content to less than $\pm 0.25\%$ and plutonium to less than $\pm 1\%$.

The standard under consideration seeks to have the calculation method used to determine uranium losses due to depletion and transmutation capable of determining the total mass of uranium at discharge to an uncertainty of no greater than $\pm 0.5\%$ at the 95% confidence level.

For all reactors there is a significant amount of information on equipment and systems that are used to determine actual reactor thermal power. The data is normally carefully taken and becomes a part of the operating

history of the reactor. The amount of it is likely to increase with the adoption of the quality assurance programs recently imposed on the reactor operators by the Atomic Energy Commission. This data can be used in the nuclear materials safeguards program.

One utility concern is the performance of large equipment and the method of monitoring the performance. Reactor operators have required calibration tests by independent testing laboratories on some of the monitors of key parameters such as flow nozzles in large pipe sections to confirm the flow data provided by the manufacturer. Reactor suppliers have mocked up pumps to give performance data including output and efficiencies. In every installation, part of the data available to the operator, who is normally working with the reactor supplier, are the results from preoperational and cold-flow tests. These are followed by the low-power tests and then the step-wise escalation to full power. During each of these steps, it is standard practice to run a test on determination of reactor core thermal power. Normally, the thermal power determination tests require several hours per test. Backup instrumentation described in PTC - 32.1 is used to further check the installed equipment. With the recent adoption of PTC - 32.1 all thermal power tests will be conducted in a standard manner, and the required data logged on standard format.

The utility concern for large equipment performance will result in the development of data and testing methods, which should influence the accuracy to which certain parameters which are required for measurement of core thermal power are known. This, in turn, will have a positive effect on the nuclear materials safeguards program. Particularly, this will affect the capability to calculate core isotopic inventory as a function of time.

B. Instrumentation

Increasing attention is being given to evaluation of burnup using added in-core instrumentation and on-line computers which can, with proper software, estimate in great detail fuel burnup. New reactors are currently coming on line with such capabilities. For example, a BWR with 164 in-core fission counters in each core and a PWR with 180 in-core fission counters are being built. In one instance, a large on-line computer is available to follow fuel burnup. Table III⁽⁷⁾ shows the uncertainty which can be attributed to important measured parameters in a typical light water reactor.

TABLE III

ESTIMATE OF BASIC UNCERTAINTY IN NUCLEAR FUEL
PERFORMANCE IN A TYPICAL LIGHT-WATER REACTOR

<u>Parameter</u>	<u>Uncertainty (No. of Standard Deviations)</u>
Reactor feedwater flow	1
Reactor recirculation flow	2
Reactor feedwater enthalpy	<< 1
Fuel assembly power	5
Relative axial power factor	5
Local peak rod power factor	3
Reactor axial water data	4
Reactor in-core detectors	6
Fuel assembly flow	3
Total reactor power	1.5
Reactor power distribution	6
Fuel assembly burnup	6-10

Table IV⁽⁸⁾ displays estimated errors associated with certain nuclear parameters in an Operating PWR.

TABLE IV
ESTIMATED ERRORS IN SOME MEASURED
PARAMETERS IN AN OPERATING PWR

<u>Parameter</u>	<u>Estimated Error</u>
Critical Boron Concentrations	± 2 PPM
Control Rod Worths	$\pm 4\%$ (Zero Power) $\pm 10\%$ (Full Power)
Moderator Temperature Coefficients	$\pm 1 \times 10^{-5}\%$ F (Zero Power) $\pm 2 \times 10^{-5}\%$ F (Full Power)
Differential Boron Worths	$\pm 5\%$ (Zero Power)
Power Coefficients	$\pm 10\%$ (Full Power)
Relative Fuel Assembly Power	$\pm 7\%$ (Zero Power) $\pm 5\%$ (Full Power)
Nuclear Hot Channel Factors	$\pm 10\%$ (Zero Power) $\pm 8\%$ (Full Power)

These two tables allow one to make some judgement as to the probable accuracy of data from power reactor cores. In particular, one can see that major advances are required in power reactor instrumentation in order to accumulate data from power reactors which will be useful to the nuclear safeguards role. For example, if the relative fuel assembly power could be measured to 3%, the accuracy for U^{235} content prediction could be almost halved.

Unusual conditions have existed which have affected the measurement of reactor thermal power and which if not detected would compromise the determination of fuel burnup. For example, temperature detectors gave erroneous data because they were improperly located and the thermal power

calculations based on these values gave obviously poor results until the source of the error was isolated and corrected. The long-term solution to this problem is better design on subsequent units.^(9,10,11) In another instance, a part of a control rod was unknowingly left in the reactor for several operating weeks. The exposure on the fuel surrounding the inserted control rod required back fitting to properly assess the burnup.

Instrumentation typically used on power reactors can be classified as operational type instrumentation. That is, it can conform to the following criteria:

- Capable of remote operation for reasonable periods of time without maintenance
- Capable of operating in hostile environments, i.e., radiation, heat, dust, etc.
- Reasonable unit costs, in terms of capital investment and maintenance.

The above factors indicate that typical operational instrumentation tends to be durable rather than extremely accurate.

There is also the primary mission to be considered. The data required to operate a power reactor in a safe and economical manner is such that extreme accuracy is not required. Therefore, operational process instrumentation tends to fulfill its primary mission of providing data with reasonable error bounds⁽¹²⁾ i.e., $\pm 4\%$, while at the same time being relatively rugged.

However, certain parameters of interest to the nuclear material safeguards program have to be measured to accuracies beyond the capabilities of standard operational instrumentation. As an example of this situation,

consider the problem of doing an accurate thermal heat balance on a power reactor for the purposes of calibrating power level nuclear instrumentation. Perhaps as part of this balance, one might be required to know the temperature drop across a heat exchanger. One can find that the temperature gauge, for taking this data, is a large needle gauge graded in steps of 10°F . This type of gauge will, of course, provide data of sufficient accuracy to insure the safe and efficient operation of the power plant. However, at the same time, this type of instrumentation can introduce significant errors in the prediction of total plutonium production.

Therefore, in the absence of special instrumentation designed and installed in power reactors for nuclear safeguards purposes, there are inherent errors in the use of operational instrumentation for the gathering of data which will be used for the nuclear materials safeguards mission.

The nuclear materials safeguards system, if it is to use operational data from power reactors as a means of safeguarding nuclear materials, has to be designed to recognize not only that operational instrumentation is conceived for a different mission than safeguards, but also that advanced techniques used to refine the accuracy of experimental instrumentation are not used in operational instrumentation.

C. Calculational Limitations

The physics behavior of a power reactor presents a complex three-dimensional problem. The analysis is complicated by perturbations such as control rods and water gaps, as well as by the effects of thermal-hydraulics. The central problem, from a fuel management point-of-view, is that of

determining the burnup, i.e., the change in fuel isotopic composition as a function of irradiation.

Fuel burnup data is dependent upon (a) an accurate determination of the integrated thermal power produced by the core, (b) proper assignment of burnup to each fuel assembly, and (c) a proper computer code which uses this information to calculate the depletion of the uranium and the buildup of the plutonium. In addition one needs to know the weight of the ^{238}U and ^{235}U in detail, certainly in each fuel assembly if not in each fuel rod in an assembly. Contracts have been written in the past in which the fuel supplier will certify the uranium content to $\pm 0.43\%$ to $\pm 0.7\%$ at the 95% confidence level for any fuel assembly and to be within $\pm 0.06\%$ to $\pm 0.1\%$ on the total weight of U in all assemblies. Other contracts have been arranged in which the uranium content of each rod in an assembly was determined to 0.1% or less. (The ^{235}U content that is given to the operator is based on an enrichment furnished to the fuel vendor by the USAEC and is normally not questioned). As mentioned earlier, proposed engineering codes will standardize this data.

Calculational methods used for predicting fuel burnup can be classified as 1) cell methods (or point burnup methods) or 2) macroscopic methods. Cell methods are used to evaluate burnup within individual fuel elements, and macroscopic methods are used to perform over-all reactor calculations using the results of cell burnup calculations. The cell methods are useful in themselves for predicting changes in reaction rates and fuel composition for a cell which is typical of a given portion of a reactor core. However, the ultimate goal of burnup calculations is to be able to predict fuel burnup over the whole reactor.

A software program typically is supplied by the reactor vendor for determining burnup and the buildup of plutonium in each fuel assembly, essentially on a continuous basis. The code is usually proprietary and the operators in several instances are developing their own to better manage their fuel inventory. The measure of success in predicting burnup is made from the comparison with the analytical data during the dissolution of the fuel. For example, comparisons with data from one campaign with preliminary estimates of burnup show agreement to within 8% in the worst case and generally within 5% at the 95% confidence level.

It is in the area of macroscopic reactor calculations that one faces the most severe limitations. The complexity of the physical problem involved is such that an exact solution is beyond the capabilities of today's large computers. Because of limitations in computer capacity, one is limited in the degree of detail that can be included in calculational models. The heterogeneity of the reactor lattice produces local spatial variations in the neutron spectrum that can have important effects on local power. Among these important heterogeneities are water gaps, control rods, non-uniform enrichment, and burnable poisons. The thermal-hydraulic effects also complicate the picture. One is thus forced to make certain simplifying assumptions to arrive at a reasonable model for the actual situation. Simplifications come into the model first of all in the basic theory, i.e., in description of the physical processes that are taking place. Further simplifications are made in describing the complex geometrical detail of the reactor core.

The general method used for evaluating water reactor lattices consists in using multigroup transport theory codes to generate homogenized few-

group constants for typical regions of the lattice, and then using these few group constants in diffusion-theory calculations of the spatial power distribution. The diffusion theory calculations are performed in either one, two, or three space dimensions. Increasing the spatial detail is done at the expense of energy detail.

Detailed two- and three-dimensional burnup calculations are very lengthy and expensive. Recently synthesis methods have been used with some success to save computer time.

Although present-day methods of calculating power distributions and burnups are considered "adequate" to permit the safe operation of reactors, they contain many recognized limitations and inaccuracies. In an assessment of the current status (1968) of reactor calculations made by the USAEC and an industrial task force, it was stated⁽¹³⁾ that "nuclear core design is still evolving. Much remains to be learned from reactors not yet operating, regarding prediction and control of the distribution of power between and along the fuel rods making up a core." It was also stated in the same report that "simplifying approximations need to be made to allow lifetime design calculations to be performed, and the extent to which these represent reality has not been firmly established."

In a summary of an IAEA-sponsored panel held in Vienna in 1967 it was stated that "there was general agreement that there is a faster-than-linear increase in the difficulty of over-all reactor analysis as core size is increased. This is a result of the combined requirements for a greater number of channels or mesh points, numerical difficulties introduced by large, loosely coupled cores, and the greater sensitivity of the power shape to small errors in local reactivity."

An additional factor to be considered in burnup calculations, and especially if one is recycling plutonium, is the fact that the reactor physics analysis is complicated if there is a large amount of plutonium in the lattice. These complications come about because of the strongly non- $1/v$ behavior of the thermal neutron cross sections of the important plutonium isotopes, ^{239}Pu , ^{240}Pu , ^{241}Pu . The thermal flux in the fuel is strongly depressed at energies near 0.3 eV and 1 eV due to large Pu resonances at these energies. Also, the uncertainties in the basic nuclear data for the plutonium-isotopes are greater than for uranium.

To summarize this brief evaluation of methods being used in burnup calculations, one can conclude that methods representing the current state-of-the-art still contain many approximations and limitations. As the capacity and speed of computers increase in future years, one can expect to see significant improvements made in the tools used to predict burnup in thermal reactors.

D. Theory-Experiment Correlations

In the previous section some of the limitations in methods currently used to predict power distributions and burnup in thermal reactors were discussed. The ultimate test of any calculational method is, of course, how well it can predict reality. This section examines the current state-of-the-art in this light by considering theory-experiment correlations.

The current state-of-the-art is still fairly well depicted in the Proceedings of a Panel on Fuel Burnup Predictions in Thermal Reactors,⁽¹⁴⁾ held in Vienna in 1967. This section draws heavily from the conclusions of that panel. In the summary section of this panel report, it was pointed out that there was considerable disagreement concerning the absolute magnitude of errors in current predictions. It was also pointed out that when com-

paring results of various burnup studies one must be careful to distinguish between a carefully-controlled, base-loaded power reactor experiment, and actual operation of a nuclear power plant on a utility grid under load-following conditions. The latter is, of course, by far the more complex situation. Also, one must distinguish between prediction of measured isotopic concentrations at a few selected points in a fuel element where burnup also has been measured, and prediction of isotopic concentrations and burnup in an overall sense over the whole reactor. Again, the latter case is by far the more complex, and the accuracy of overall predictions depends very strongly on the accuracy of predicted spatial distributions of power.

While there is sufficient information on the theory-experiment correlations of uranium cores, the plutonium recycle data has not been consistently evaluated at this time nor is there as much data available as one might like. On the basis of the much more complex energy dependence of the plutonium cross sections, one would expect the a-priori correlation to exhibit more errors than in the uranium case. There will be a Vienna panel on plutonium recycle in 1971 which should help to establish the consistent evaluation of the available data. From this evaluation, one could expect to then derive the state-of-the-art for plutonium recycle cores.

In a discussion of theory-experiment discrepancies one cannot assign the total discrepancy to inaccuracies in theory, but must also take into account the inherent uncertainties in measured quantities. For quantities measured in an operating power reactor, experimental uncertainties can be significant.

The Vienna panel of 1967 attempted to put their summary of the state-of-the-art of burnup physics on a quantitative basis by commenting on experimental uncertainties as well as on accuracy of predictions.

Regarding flux and power distributions, a general comment was made⁽¹⁵⁾ that "the precision achievable in estimating many burnup quantities is severely limited by knowledge of the absolute thermal power level of a reactor." Process instrumentation was cited to be typically uncertain to $\sim 4\%$ of the total thermal output of the plant. The accuracy of in-core instrumentation was cited as about $\pm 5\%$. The precision of power distribution predictions in light water reactors was cited as 3-5% (r.m.s. deviation) which is within instrumentation error. In general, however, deviations over the whole core fall in an error range of about 10%, and errors as large as 20% have been reported for low flux regions. Furthermore, systematic trends have been noted with region and exposure. These rather large errors are attributed to "inadequate theoretical modeling of the reactor, both as to the spatial resolution and physical description."

A typical set of values for predicting the isotopic content of fuel based on the operating history of a reactor and a computer model which distributes thermal power to the fuel assemblies correctly to within 10% yields:

^{235}U to $\pm 3.1\%$

Pu to $\pm 7\%$

^{238}U to $< 0.1\%$

^{236}U to $\pm 6\%$

Regarding isotopic buildup and depletion, it was stated by the panel that "in general, the error in predicting the ratio of ^{239}Pu content to ^{235}U destruction is about 5% of its value." As to the prediction of isotopic ratios of various plutonium isotopes, it was stated that "the error in the ratio of each higher plutonium isotope to its predecessor may be $\sim 10\%$."

A paper was presented at the Vienna Meeting by Crowther of the General Electric Company, in which he discussed the burnup analysis of large boiling water reactors. He pointed out the difficulties that are introduced into the analysis by the many heterogeneities present in a BWR. Also, because of significant spatial variations in moderator density (both axial and radial) the analysis of a BWR is definitely a three-dimensional problem.

The agreement between measured power distributions and those calculated using current methods is surprisingly good considering the approximations contained in the methods. These calculations have generally predicted power distributions to within approximately 10%, with the largest discrepancies usually occurring near the reflector or near inserted control rods. Errors in these low-flux regions have only a small effect on the ability to meet thermal limits. However, they can have important effects on the fuel burnup analysis because of the rather large volume of fuel in the reflector zones. In support of his $\pm 10\%$ figures, Crowther presented specific comparisons between coarse-mesh, three-dimensional calculations and measured power distributions from the Dresden Nuclear Power Station and from the Garigliano (SENN) Station.

Regarding theory-experiment correlations of isotopic concentrations and ratios, data were presented on the Dresden and Yankee reactors which showed the following: The mass ratio of plutonium to uranium was underestimated for both reactors - by 5% for Dresden and by 0.4% for Yankee; the fissile plutonium fraction was underestimated for both reactors - by 4.5% for Dresden and by 3.1% for Yankee. The amount of ^{241}Pu in Dresden was overestimated by 11%.

Table V⁽¹⁶⁾ shows an evaluation of Dresden isotopic composition measurements for various isotopes of interest.

TABLE V

COMPARISON WITH DRESDEN ISOTOPIC COMPOSITION MEASUREMENTS

<u>Isotope</u>	<u>Calculated Percentage Difference (Versus Measured)</u>
U^{235}	- 1.4
U^{236}	0.0
Pu^{239}	- 6.8
Pu^{240}	+ 6.7
Pu^{241}	+10.8
Pu^{242}	+ 3.8
Np^{237}	-11.0
Am^{243}	-33.0
Cm^{244}	-38.0

Discrepancies of the same order have been reported by Lacey and Radcliffe⁽¹⁷⁾ of Westinghouse. Their work represents the state-of-the-art as it stands for pressurized water reactors, and is referenced in current Westinghouse PSARs. Reference(18) presents an analysis of Yankee Core I, together with a theory-experiment comparison. The basic analytical tool used in this study was the TURBO code, a two-dimensional, few group, diffusion-theory depletion code. A simple synthesis method was used to make final estimates of material inventories, power distributions, and reactivity. The end of life mass ratio of plutonium to uranium was underestimated by 2.1%. The fissile plutonium fraction was underestimated by 2.3%. The amount of ^{241}Pu was underestimated by 11%. Table VI⁽¹⁷⁾ shows an evaluation of fuel depletion in the Yankee Power Reactor.

TABLE VI
EVALUATION OF FUEL DEPLETION AND PLUTONIUM
BUILDUP IN THE YANKEE POWER REACTOR

<u>Isotope</u>	<u>Error (Calculation Versus Measurement)</u>
U-234	$\pm 2.7\%$
U-235	$\pm 0.9\%$
U-236	± 2.9
U-238	$\pm 0.1\%$
Total U	$\pm 0.1\%$
Pu-239	$\pm 2.1\%$
Pu-240	$\pm 3.4\%$
Pu-241	$\pm 4.4\%$
Pu-242	$\pm 6.0\%$
Total Pu	$\pm 1.8\%$

Table V and VI represent data on isotopic composition measurements for two specific reactor cores. Table II in the summary represents current and future performance for reactor cores considering all of the data available. In considering current performance, the values of U^{235} , U^{236} , U^{238} and total U, the current uncertainties are taken from the Vienna panel assuming a 10% error in thermal power. The Pu^{239} error is based on the Yankee and Dresden data recognizing, however, that the Vienna panel assigns a slightly higher number. The higher plutonium isotope values are generally consistent with the Vienna panel's recommendation of the ratio of a given plutonium isotope to the next highest

plutonium isotope and the number assigned to Pu^{239} . The total plutonium number, then, is consistent with the above performance numbers and the probable makeup of the discharge fuel. Probable future performance numbers are based on the assumption that the accuracy with which one measures thermal power can be halved. The numbers, then, are generally consistent with this assumption. A future paper will deal more explicitly with that assumption.

Available good burnup data are rather sparse. On the basis of the limited information available, and the studies that have been reported, it is clear that state-of-the-art calculational methods contain uncertainties which, especially from a nuclear materials safeguards point of view, are undesirably large.

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