Comparison of Analysis Methods for Burnup Credit Applications*

M.C. Brady[1], J.P. Renier[1], C.V. Parks[1], T.L. Sanders[2]

[1]*Oak Ridge National Laboratory, Oak Ridge, Tennessee*
[2]*Sandia National Laboratories**, Albuquerque, New Mexico, United States of America*

INTRODUCTION

The current approach used for the development and certification of spent fuel storage and transport casks requires an assumption of fresh fuel isotopics in the criticality safety analysis. However, it has been shown (Sanders et al. 1987) that there is a considerable reactivity reduction when the isotopics representative of the depleted (or burned) fuel are used in a criticality analysis. Thus, by taking credit for the burned state of the fuel (i.e., burnup credit), a cask designer could achieve a significant increase in payload.

Accurate prediction of k_{eff} for spent fuel arrays depends both on the criticality safety analysis and the prediction of the spent fuel isotopics via a depletion analysis. Spent fuel isotopics can be obtained from detailed multidimensional reactor analyses, e.g. the code PDQ (Caldwell, 1967), or from point reactor burnup models. Detailed isotopics resulting from PDQ analyses have been obtained from the Virginia Power Company for Cycle 5 of the North Anna Unit 1 Reactor. The detailed isotopics are used in KENO V.a (NUREG/CR-0200 1984) models of these reactors under restart and end-of-cycle critical conditions. These reactor calculations will help verify the adequacy of the isotopics and determine Δk_{eff} biases for various analysis assumptions (with and without fission products, actinide absorbers, burnable poison rods, etc.). New software developed to interface PDQ multidimensional isotopics with KENO V.a reactor and cask models is described. A number of different axial fuel zones are used to determine the effect of the axial variation in burnup on k_{eff}. Analyses similar to those performed for the reactor cases are carried out with a representative burnup credit cask model using the North Anna fuel.

This paper presents the analysis methodology that has been developed for evaluating the physics issues associated with burnup credit. It is applicable in the

* This work performed at Sandia National Laboratories, Albuquerque, New Mexico, supported by the United States Department of Energy under Contract DE-AC04-76DP00789.

** A United States Department of Energy Facility

validation and characterization of fuel isotopics as well as in determining the influence of various analysis assumptions in terms of Δk_{eff}. The methodology is used in the calculation of reactor restart criticals and analysis of a typical burnup credit cask.

ANALYSIS METHODOLOGY

The basic methodology may be described as having three primary stages: (1) the generation of burnup-dependent isotopics (number densities), (2) the preparation of cross-section data, and (3) the criticality analysis. An interface module,CONDENSE, has been developed to facilitate the use of first-stage isotopics in the cross-section processing codes of the second stage of analysis. Each of these stages is described below as related to the overall burnup credit program. Figure 1 is a schematic drawing illustrating the overall methodology.

BURNUP-DEPENDENT ISOTOPICS

In the first stage, the isotopics (nuclide number densities) for the spent fuel are calculated from either multidimensional models or a simpler point reactor model. There are two multidimensional models being considered in the analysis. PDQ-7 is a program commonly used by vendors and utilities to solve the neutron diffusion-depletion problem in three dimensions. The GPRCYCLE computational system developed at Oak Ridge National Laboratory permits multidimensional, multizone, and multigroup reactor depletion calculations to be performed by accessing a variety of computational modules, including modules from the VENTURE (Vondy et al. 1981), SCALE (NUREG/CR-0200 1984), and AMPX (Greene et al. 1976) code systems. These multidimensional models take into account the explicit operating histories of each assembly (including fuel shuffling patterns, control rod positions, etc.), basically simulating reactor operation.

The point depletion model used in SAS2H (Shielding Analysis Sequence 2H) of the SCALE code system provides a useful tool for characterizing the spent fuel isotopics in terms of their effect on k_{eff}. SAS2H is an automated sequence that uses ORIGEN-S (Hermann and Westfall 1984) for the depletion and relies on a 1-D transport theory code to update the cross sections and neutron flux spectrum as a function of fuel burnup. Separate SAS2H analyses may be performed for different fuel assembly types (enrichment, presence of burnable poison rods, water holes, burnup, etc.). The other advantage of using point depletion models is that the effect of particular nuclides or groups of nuclides (fission products, actinide absorbers, etc.) can be evaluated in a cost-effective manner. Presently under development is an interface module, SAS2H Nuclide Inventories for CONDENSE and KENO Runs (SNICKR), which will process the isotopics generated by one or more SAS2H calculations for use in criticality analyses.

In the current stage of analysis, the isotopics used have been restricted to those obtained from Virginia Power (VP). Spatial isotopics resulting from the VP-PDQ analyses have been obtained for Cycle 5 of the North Anna Unit 1 Reactor and Cycle 2 of the Surry Unit 1 Reactor. These data include the 29 nuclides listed in Table 1 which are considered explicitly in the VP analysis. Additional fission product nuclides have been included via the use of a "lumped fission product"

(LFP). Surry Cycle 2 isotopics from GPRCYCLE will also be used in later work to provide an independent verification of the results based on VP-PDQ data.

The CONDENSE Module

The use of detailed isotopics (approximately 48,074 spatial mesh points for the North Anna data) required the development of a code to reduce the data to an acceptable level for use in criticality analyses. CONDENSE, a user-oriented code developed for this purpose, produces the data required for the three phases of the cross-section processing stage shown in Fig. 1. The basic functionality of CONDENSE may best be described as mapping. The user may define a set of depletion zones for the condensed data by specifying the number of such zones in the x-y plane and the number of axial planes to be included. Data are input to map the VP-PDQ depletion zones to the user-defined depletion zones in the x-y plane. Similarly, information describing how the original VP-PDQ mesh in the z-direction corresponds to the condensed mesh in the z-direction is also required. The number densities calculated for these condensed depletion zones will be used in the second stage to create the mixed cross sections for use in the criticality analysis.

In order to reduce the cross-section processing and permit more flexibility, CONDENSE allows the user to define cross-section zones (or batches) that are different from the depletion zones. These cross-section zones are used to produce the data required for the generation of burnup-dependent cross sections. The cross-section zones are defined in the same manner as the depletion zones, the number of cross-section zones in the x-y plane is specified as well as the number of axial planes to be used in the condensed data. Data describing how the VP-PDQ depletion zones (mesh) are to correspond to the cross-section zones specified by the user are input in the same manner as for the depletion zones. CONDENSE uses the average nuclide number densities calculated for these cross-section zones and other user-supplied data (temperatures, boron concentrations, water densities, etc.) to create input files for the SCALE cross-section processing sequence, CSASN (NUREG/CR-0200 1984). Additional information is required to relate the depletion zones with the batch cross-section zones in producing the mixed cross sections for the KENO V.a analysis. The third major function of CONDENSE is to provide data to a series of codes that are used to calculate the lumped fission product mentioned earlier. CONDENSE identifies the fission product nuclides whose number densities are explicitly given in the detailed isotopics and passes that information on to the second stage of the analysis.

Cross-section Analysis

As described above, CONDENSE provides the data required by each of the three phases in the production of cross-section data for the KENO V.a analysis. Basic cross-section data used in these analyses are from the SCALE 27-neutron-group cross-section library which is based on ENDF/B-IV enhanced with fission-product data from ENDF/B-V. This library has been validated against low-enriched pin-type lattice critical experiments. The calculation of the batch cross-section data is fairly automated. CONDENSE provides input data for each cross-section batch which is used in a CSASN calculation to produce appropriate cross-section data. The calculation of the LFP is less straightforward. The data from CONDENSE are used in conjunction with isotopics produced from a SAS2H calculation as input to

the LUMPIT routine which produces cross-section data for a LFP comprised of approximately 200 fission products less those explicitly contained in the VP-PDQ analysis. The cross-section data are combined into a single dataset and are then used with the depletion zone number densities produced by CONDENSE in the form of ICE (NUREG/CR-0200 1984) input to produce the final mixed cross sections required in the KENO V.a analysis.

Criticality Analysis

In this final stage of the analysis, KENO V.a is used to model either the reactor (used in the validation and characterization of isotopics) or a burnup credit cask (used in the characterization of isotopics and parameterization of k_{eff}). Generally, the KENO models are highly detailed, comprised of individual fuel pins. Detailed KENO V.a models for the North Anna Unit 1 Reactor have been developed. In the case of the cask model, the amount of detail required by the KENO model (in terms of the number of different assemblies that must be modeled) depends on the number of cross-section batches in the condensed data that are to be used in the cask. The current KENO cask model is for a 31-element cast-iron cask with a borated stainless steel basket. As with the reactor models, each type cask analyzed will require its own KENO model.

SCOPE OF ANALYSIS

Adequately characterizing the requirements for spent fuel isotopics to be used in evaluating burnup credit in spent fuel casks requires the validation of the general analysis methods to be employed and identification of potential sources of bias that can result from various modeling assumptions as well as physical uncertainties about the fuel itself.

Reactor Models

The methodology described earlier is applied to reactor restart criticals in order to meet the requirement of ANSI/ANS 8.1 that recommends method validation against critical experiments. Data received from VP and the analyses performed for the North Anna Unit 1 reactor were for conditions of hot zero power (HZP) at the beginning-of-cycle (BOC) 5. In both cases all regulating rods are out (ARO). These reactor criticals are also used in quantifying the influence of different modeling assumptions in terms of changes in k_{eff}. The presence of burnable poison rods (BPR), soluble boron, and fission products (FP) are examined. Bias in k_{eff} as a function of the number of axial zones used in the analysis is evaluated for 1, 3, 7, and 22 axial zones.

Cask Model

A generic 31-element burnup-credit cask was modeled in order to identify potential sources of bias in cask calculations. The variations in k_{eff} as a function of modeling assumptions and physical uncertainties were explored. These modeling assumptions were similar to those used in the reactor analysis; the presence of BPRs during depletion, the lumped fission product used in the VP analysis, and the variation in k_{eff} with the number of axial zones. It has been noted that there is enough reactivity

in one to two feet of fresh fuel to cause a moderated array of assemblies to potentially become critical. A study modeling a cask loaded with burned fuel assemblies in which 1 foot of fuel on each end of the assembly has been replaced with fresh fuel isotopics is planned in order to evaluate the significance of this concern.

RESULTS

Reactor Study Results: Preliminary analyses of the North Anna Unit 1 reactor under the conditions of HZP at BOC-5 using the VP-PDQ isotopics have been completed. North Anna uses 17x17 Westinghouse assemblies with 24 guide tubes and one instrument tube per assembly. There are 157 assemblies in the core. Batch dependent theoretical densities were used for the UO_2 pellets whose outer diameter is 0.81915 cm. The fuel rod cladding is Zircaloy-4 with a thickness of 0.05715 cm and a diametrical gap of 0.00825 cm was used. The burnable poison rods (BPR) are of borosilicate glass. The initial enrichment of the fuel ranged from 3.21 to 3.60%. Burnups for the fuel assemblies in-core at BOC-5 ranged from 0.0 GWd/MTU (fresh fuel) to 31.5 GWd/MTU. All regulating banks were withdrawn and the soluble boron concentration was 1650 ppm. The parameters that were varied to determine their influence on k_{eff} were the presence of BPR's, soluble boron, and fission products (FP), the use of spent and/or fresh fuel and the number of axial depletion zones. Table 2 summarizes the results for the North Anna fuel.

The first row of Table 2 describes the HZP conditions and is the case used to validate the methodology. The one-axial-plane result for this case is within 1% of critical. Twenty-two (22) axial depletion zones were also used to calculate the HZP conditions with a resultant k_{eff} of 0.9951 ± 0.0028. Three axial depletion zones appear to give the best result. The relatively small change in k_{eff} as a function of the number of axial depletion zones may be because the VP-PDQ data was beginning-of-cycle data and 59 of the 157 fuel assemblies in the core were fresh fuel. The case of depleted fuel in the cask is expected to show more axial variation, although the differences seen for the reactor critical are greater than one standard deviation.

Cask Study Results: Calculations to determine the sensitivity of k_{eff} on the number of axial depletion zones have also been made using a detailed model of a 31-element burnup-credit spent-fuel cask. In these initial studies, the cask was modeled as containing North Anna Unit 1 spent fuel assemblies with an average burnup 31.5 GWd/MTU (using VP-PDQ isotopics). One, three, and seven axial depletion zones were used in these calculations resulting in k_{eff} values of 0.8583 ± 0.0013, 0.8613 ± 0.0012, and 0.8684 ± 0.0012, respectively.

The only parametric study performed using the cask model has been to quantify the effect of the lumped fission product (LFP) used in the VP analysis. The model cask was loaded with 31 assemblies of 31.5 GWd/MTU fuel and then with assemblies burned to 20.9 GWd/MTU. Calculations with and without the LFP were performed for each of these cask loadings. Differences of 3.9% for the 31.5 GWd/MTU loaded cask and 2.6% for the lower burned fuel were calculated. These differences may not be representative of that expected for a typical cask loading. Recall that the isotopics used in this analysis are based on BOC data and that the Waste Policy Act requires fuel to be cooled at least five years prior to transport. Also, the

magnitudes of these numbers are highly dependent on which nuclides are included in the lumped fission product. Earlier reports (Sanders et al. 1987 and Parks 1988) have identified and ranked nuclides considered important in the characterization of LWR fuel. Specifically, 37 nuclides have been identified for inclusion in criticality analyses (Parks 1988). The VP-PDQ analysis only contains 17 of these 37. If the SAS2H sequence is used to calculate number densities, these effects can be separated out and quantified more precisely.

DISCUSSION

The majority of the work to this point has been in developing the analysis methodology and the interface module, CONDENSE. The results presented for the North Anna Unit 1 reactor validate the methodology described in this paper. Results based on the various anlaysis assumptions in both the reactor and cask studies have been useful in defining specific areas for additional study. The use of the SAS2H sequence to calculate isotopics based on point depletion will allow more flexibility in characterizing potential sources of bias in k_{eff} and establishing a procedure to quantify an acceptable Δk margin for a particular cask. Fission-product nuclides that have been identified as important to the characterization of LWR fuel (Sanders et al. 1987 and Parks 1988) will be included explicitly and their influence on k_{eff} quantified. Hopefully, this will decrease the importance of the lumped fission product that was observed in the cask results. An additional advantage of using the SAS2H isotopics is that they will be more representative of fuel that is to be shipped in the cask.

ACKNOWLEDGMENTS

The authors wish to express their appreciation to Dr. J. R. White of the University of Lowell for his work in the early stages of development of CONDENSE. The cooperation and assistance of M. Smith and S. Ahmed of the Virginia Power Company in providing the PDQ isotopics used in this paper are also gratefully acknowledged.

REFERENCES

Cadwell, W.R., "PDQ--7 Reference Manual", WAPD-TM-678, Westinghouse Advanced Power Division (1967).

Greene, N.M., et al., "AMPX: A Modular Code System for Generating Coupled Neutron-Gamma Libraries from ENDF/B", ORNL/TM-3706, Union Carbide Corp., Nuclear Division, Oak Ridge Natl. Lab. (1976)

Hermann, O.W., and Westfall, R.M., "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source terms," as described in Sect. F7 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. 1-3, NUREG/CR-0200, U.S. Nuclear Regulatory Commission (originally issued July 1980, reissued January 1982, Revision 1 issued July 1982, Revision 2 issued June 1983, Revision 3 issued December 1984).

Parks, C.V., "Parametric Neutronic Analyses Related to Burnup Credit Cask Design," to be published in Proceedings of Workshop on Burnup Credit Issues in Spent Fuel Transportation, February 1988, Washington, D.C.

Sanders, T.L., Westfall, R.M., and Jones, R.H., "Feasibility and Incentives for the Consideration of Spent Fuel Operating Histories in the Criticality Analysis of Spent Fuel Shipping Casks", SAND87-0157, TTC-0713, Sandia National Laboratories (August 1987).

SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. 1--3, NUREG/CR-0200, U.S. Nuclear Regulatory Commission (originally issued July 1980, reissued January 1982, Revision 1 issued July 1982, Revision 2 issued June 1983, Revision 3 issued December 1984).

Vondy, D.R., Fowler, T.B., and Cunningham, G.W.,"The Bold VENTURE Computation System for Nuclear Reactor Core Analysis, Version III", ORNL-5711, Union Carbide Corporation, Nuclear Division, Oak Ridge Natl. Lab. (1981)

Table 1. Nuclides Explicitly Contained in VP-PDQ Analysis

U-234	Ru-103
U-235	Rh-103
U-236	Rh-105
U-238	I-135
Pu-239	Xe-135
Pu-240	Nd-147
Pu-241	Nd-148
Pu-242	Pm-147
Am-241	Pm-148
O-16*	Pm-148
H-1*	Sm-149
B-10*	Eu-153
B-11*	Eu-154
Zirc-4*	Eu-155
	Gd-155

*Number densities used in the criticality analyses for these nuclides are calculated seperately

Table 2. Preliminary k_{eff} Results[a] for North Anna Unit 1, BOC5

Parameters included in analysis					Number of axial depletion zones	
Spent fuel	Fresh fuel	BRP	Soluble Boron	FP	1	7
X	X	X	X	X	0.9907(27)[b]	0.9932(28)
X	X	X	X			1.0259(28)
X	X	X		X		1.1437(31)
X	X		X	X		1.0521(26)

[a]Based on 153 generations, 300 neutrons/generation, 32 generations skipped
[b]Read as 0.9907±0.0027.

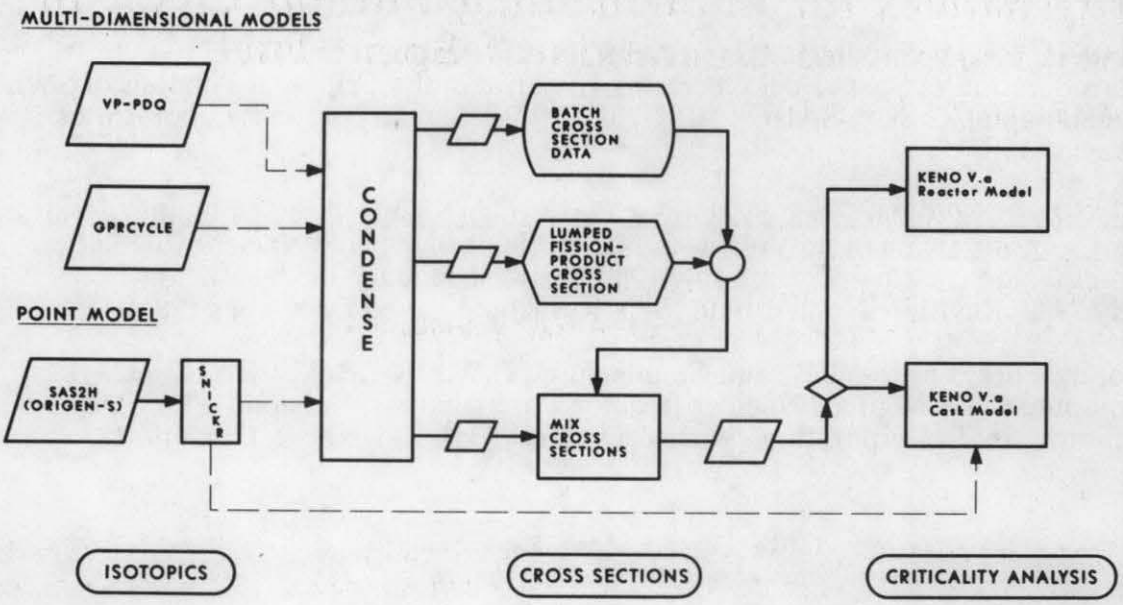

Figure 1. Schematic of Analysis Methodology