SUMMARY OF CHANGES TO
NEUTRONICS DESIGN METHODS AND VERIFICATION
OPPD-NA-8302-NP
Rev. 04

1.	Title Page	Change the revision number and date.
2.	All Pages	Update the revision number.
3.	Pages iii-iv	Update Table of Contents with new page numbers and - Replace ABB/CE codes (ROCS, MC, HERMITE) with Studsvik codes (CASMO-3/SIMULATE-3), ESCORE and ASAS. - Revise Section 1.0 through 5.0 to account for new code descriptions, models and benchmarking.
4.	Page v	Update List of Tables.
5.	Pages vi-vii	Update List of Figures.
6.	Page viii	Update revision sheet.
7.	Pages 1-2	Update Introduction section: - Introduce use of new codes. - Brief description of similar NRC-approved methods used by Yankee Atomic Electric Company. - Primary objective of methodology revisions. - Brief description of each major section. - Table of key physics parameters included in benchmark.
8.	Pages 3-11	Revised Methods Description section: - Description of new codes (ESCORE, CASMO-3, CASLIB, TABLES-3, SIMULATE-3 and ASAS). - Description of computer models for fuel assemblies and reflectors (CASMO-3), cross-section tables (TABLES-3), neutronics depletions (SIMULATE-3) and axial shape analysis (ASAS). - Figure for new code(s) sequencing and flow chart.
9.	Pages 12-18	Revisions to Application of Physics Methods: - Replace "ROCS" or "MC" with "SIMULATE-3" where appropriate. - Replace "hot zero power" with "HZP" where appropriate. - Insert new method for determining MTC and FTC. - Insert new method for calculating reactor kinetics. - Insert new method for determining the appropriate axial shape data.
10.	Pages 19-22	Include new section describing the benchmark of the CASMO-3/SIMULATE-3 models for Fort Calhoun Station. - Description of measurement data base. - Description of comparison of predictions to measurements. - Summary of applicable tables and figures. - Conclusion of each comparison. - Summary of benchmark effort.

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11. Page 23 — Include Conclusions section describing the overall results of the benchmarking project and the ability of OPPD personnel to perform neutronics calculations used for future FCS core reload analyses.

12. Pages 24-26 — Update References section to include applicable Studsvik code manuals, YAEC topicals, B&W reports for critical experiments and updated revisions to existing OPPD reload methodologies.

13. Pages 27-81 — Prepared tables and figures for the CASMO-3/SIMULATE-3 benchmark:
   - ROCS-DIT predictions, SIMULATE-3 predictions and zero and full power test measurements compared for CBCs, ITCs, PCs and CEA Group Worths.
   - SIMULATE-3 predictions and core follow measurements compared for at-power CBCs, radial powers in incore instrumented boxes, normalized axial power shapes and pin peaking factors.

Omaha Public Power District
Nuclear Analysis
Reload Core Analysis Methodology

NEUTRONICS DESIGN METHODS AND VERIFICATION

OPPD-NA-8302-NP
Rev. 04

May 1994

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ABSTRACT

This document is a Topical Report describing the Omaha Public Power District (OPPD) reload core neutronics design methods for application to Fort Calhoun Station Unit No. 1.

The report addresses OPPD's neutronics design methodology and its application to the calculation of specific physics parameters for reload cores. In addition, comparisons of results obtained using this methodology to results from experimental measurements and independent calculations are provided.

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00	September 1983
01	November 1986
02	April 1988
03	January 1993
04	May 1994

OMAHA PUBLIC POWER DISTRICT
RELOAD CORE ANALYSIS METHODOLOGY
NEUTRONICS DESIGN METHODS AND VERIFICATION

1.0 INTRODUCTION

This document describes the Omaha Public Power District (OPPD) neutronics design calculation methods using the CASMO-3/SIMULATE-3 computer code system. Studsvik AB and Studsvik of America (SOA) developed the CASMO-3/SIMULATE-3 computer code system which is considered a state-of-the-art core analysis modelling system throughout the nuclear industry (References 1-1 through 1-4).

Previous Fort Calhoun Station (FCS) neutronics design methods approved by the NRC were based on ABB/Combustion Engineering (ABB/CE) topical reports. The FCS neutronics design methods described in this document are based upon a combination of NRC-approved methods from ABB/CE topical reports and NRC-approved Yankee Atomic Electric Company (YAEC) topical reports involving the use of CASMO-3/SIMULATE-3. YAEC previously provided the theoretical basis and validation of the CASMO-3/SIMULATE-3 computer code system to the NRC (References 1-5 and 1-6). These topical reports provided detailed descriptions of the computer programs and a general methodology for performing reactor physics analyses.

The primary objective of this methodology report revision is to demonstrate OPPD's ability to use the CASMO-3/SIMULATE-3 computer code system to accurately model the FCS reactor for the purposes of core reload design analysis. Table 1-1 lists the reactor physics parameters used for benchmarking CASMO-3/SIMULATE-3 predictions against plant measurement data.

Section 2.0 provides a basic description of physics methods and models. Section 3.0 details OPPD's application of these models to the FCS reactor. Section 4.0 discusses OPPD's latest verification program that includes the cycle-by-cycle comparisons of OPPD calculated data to measured data and data from independent calculations using the CASMO-3/SIMULATE-3 computer code system. Section 5.0 contains the overall conclusions. Section 6.0 lists the individual references.

TABLE 1-1

REACTOR PHYSICS PARAMETERS BENCHMARKED USING CASMO-3/SIMULATE-3

- Critical Boron Concentration (Core Reactivity)
- Power Coefficient
- Isothermal Temperature Coefficient
- Control Rod Worth
- Assembly Power Distributions
- Assembly Pin Peaking Factor
  - Integrated Radial Peaking ($F_R/F\Delta h$*)
  - Planar Radial Peaking (F_{XY})
  - 3-D Peaking (F_Q)

* Integrated Radial Peaking output from SIMULATE-3 is designated as $F\Delta h$.

2.0 DESCRIPTION OF PHYSICS METHODS AND MODELS

OPPD's neutronics design analysis for the FCS core using CASMO-3/SIMULATE-3 is based on the continuing effort to improve and enhance OPPD's capabilities in performing reload design calculations. The methodology employs a similar combination of multi-group neutron spectrum calculations used in the currently approved methodology. Use of the CASMO-3 provides cross-sections appropriately averaged over a few broad energy groups. SIMULATE-3 provides two-group, two- and three-dimensional diffusion theory calculations which result in integral and differential reactivity effects and power distributions. These programs embody more advanced analytical procedures than the currently approved methods and use the fundamental nuclear data consistent with the current state-of-the-art technology.

2.1 DESCRIPTION OF COMPUTER PROGRAMS

The CASMO-3/SIMULATE-3 computer program system was developed by Studsvik AB, Nykoping, Sweden, and their American subsidiary Studsvik of America, Newton, Massachusetts. The computer program package consists of the following computer programs:

- CASLIB
- CASMO-3
- TABLES-3
- SIMULATE-3

In addition, the Electric Power Research Institute's (EPRI) ESCORE computer program was incorporated into the new methods along with the ASAS computer program jointly developed by Yankee Atomic Electric Company (YAEC) and OPPD.

2.1.1 ESCORE

ESCORE (Reference 2-1) is a computer program that predicts best-estimate, steady-state fuel performance data for light water reactor fuel rods. This computer program previously received NRC approval for use in calculating fuel rod temperatures for input to design and safety analyses (Reference 2-2). OPPD uses ESCORE to calculate the fuel temperature of the average rod as a function of burnup. Output from this computer program provides the average fuel pin temperature for use in CASMO-3 and a burnup dependent fuel pin temperature for the FCS SIMULATE-3 model.

2.1.2 CASLIB

CASLIB (Reference 2-3) produces a binary neutron cross-section library for input to CASMO-3 from a card-image, formatted library. This library is based mainly on data from ENDF/B-IV with an update from ENDF/B-V and other sources. Both 40- and 70-group cross-section data are available for nearly 100 materials.

2.1.3 CASMO-3

CASMO-3 (Reference 2-4) is a multi-group, two-dimensional transport theory computer program. This computer program models cylindrical fuel rods of varying composition in a square pitch array. CASMO-3 can model fuel rods, fuel rods with an integral burnable absorber material, burnable absorber rods, control rods, CEA guide tubes, in-core instruments and water gaps.

CASMO-3 generates all cross-section data for SIMULATE-3. OPPD uses CASMO-3 in a single assembly format with reflective boundary conditions and a 40-energy group cross-section library.

2.1.4 TABLES-3

TABLES-3 (Reference 2-5) is a data processing program that links CASMO-3 to SIMULATE-3. The program processes two-group cross-sections, discontinuity factors, fission product data, in-core instrument response data, pin power reconstruction data, and kinetics data from CASMO-3. TABLES-3 reads the CASMO-3 card image files and produces a master binary cross-section library for SIMULATE-3.

2.1.5 SIMULATE-3

SIMULATE-3 (Reference 2-6) is a two- or three-dimensional (2-D or 3-D), two-group coarse mesh diffusion theory reactor simulator program. The program explicitly models the baffle/reflector region, thus eliminating the need to normalize to higher-order fine mesh calculations such as PDQ. Homogenized cross-sections and discontinuity factors are applied to the coarse mesh nodal model to solve the two-group diffusion equation using the QPANDA neutronics model. The QPANDA model is the spatial neutronics model used in SIMULATE-3 which solves the three-dimensional, two-group neutron diffusion equation using fourth order polynomials to represent the intra-nodal flux distributions in both the fast and thermal groups. QPANDA explicitly treats group-to-group coupling effects on the intra-nodal flux distributions, an important phenomenon which is ignored in conventional nodal models.

The nodal thermal hydraulic properties are calculated based on the inlet temperature, RCS pressure, coolant mass flow rate, and the heat addition along the channels.

The pin-by-pin power distributions, on a 2-D or 3-D basis, are constructed from the inter- and intra-assembly information from the coarse mesh solution and the pin-wise assembly power distribution from CASMO-3.

The SIMULATE-3 program performs a macroscopic depletion. Individual uranium, plutonium and lumped fission product isotope concentrations are not computed. However, microscopic depletion of iodine, xenon, promethium and samarium is included to model typical reactor transients.

2.1.6 ASAS

ASAS (Axial Shape Analyzer for SIMULATE-3, Reference 2-7) is a computer program that formats SIMULATE-3 power distribution data into axial shape data to be used as input to the setpoint analyses. ASAS processes 3-D SIMULATE-3 power distributions and writes selected information to a user output file, a Core Transient Summary (CTS) file and an Axial Transient Summary (ATS) file.

2.2 DESCRIPTION OF COMPUTER MODELS

OPPD uses the reactor physics computer programs described in Sections 2.0 and 2.1 to model the FCS reactor core. The computer codes that embody these basic physics models are maintained on the OPPD Nuclear Engineering Workstation Network. OPPD maintains all documentation and quality assurance programs related to these workstation computer codes. The following paragraphs discuss the specifics of the FCS models.

2.2.1 CASMO-3 FUEL ASSEMBLY AND REFLECTOR MODELS

Each unique PWR fuel assembly type (defined by geometry, enrichment and burnable poison pins) is separately modelled in CASMO-3 using octant geometry. Enrichment zoning among fuel pins, burnable poison pins and CEA guide tubes are explicitly modelled. The water gap between assemblies in the reactor core is included in the CASMO-3 model. The spacer grids are also included. Design bases documents, such as the Updated Safety Analysis Report (USAR), reload reports, and as-built drawings provide the necessary data to develop the CASMO-3 assembly models.

Three depletion cases are needed to generate the average cross-section data for each fuel assembly type. First, the fuel assembly is depleted at hot full power (HFP), reactor average conditions. Moderator temperature, fuel temperature and soluble boron concentration are set to constant average values for the complete depletion. The average fuel temperature at HFP conditions is calculated with ESCORE (Reference 2-1). Second, the fuel assembly is depleted at a low moderator temperature corresponding to hot zero power (HZP) conditions. The fuel temperature and the soluble boron concentration, however, are kept at the constant HFP, reactor average values. In the final depletion, the fuel assembly is again depleted at constant HFP, reactor average conditions, but with a constant soluble boron concentration higher than is usually seen in normal operation. Restart calculations are performed from the base depletion to modify parameters from the base case. Each fuel assembly type is depleted to 70 GWD/MTU assembly average burnup using the CASMO-3 default depletion steps.

Branch cases are performed to calculate instantaneous effects. The instantaneous effects are individually calculated and used to create the proper fuel assembly cross-sections. The branch cases are executed from the HFP, reactor average condition restart files at 0, 1, 3, 5, 7, 10, 15, 20, 30, 40, 50, 60 and 70 GWD/MTU. Branch cases are run for off-normal moderator temperatures, fuel temperatures, soluble boron concentrations, control rod insertions and shutdown cooling effects.

Cross-sections to handle cold conditions (<532°F) are also generated with temperature points at 515°F, 350°F, 210°F and 68°F. The cross-sections generated in this manner utilize off-nominal and restart cases in a similar way to the normal cross-sections. The cold cross-sections are used to calculate reactivity parameters at cold conditions.

CASMO-3 also generates top, bottom and radial reflector cross-sections. Two radial reflectors were modelled to account for the varying distances between the core shroud and the core support barrel. The first radial reflector is a combination of the core shroud, a homogenous mixture of water and stainless steel centering plates located between the core shroud and the core support barrel, and the core support barrel. The second radial reflector consists of the stainless steel core shroud followed by about 15 centimeters (cm) of the homogenous mixture of water and stainless steel centering plates. The top reflector extends from the top of the active fuel to the lower surface of the fuel assembly upper end fitting. The bottom reflector extends from the bottom of the active fuel to the lower surface of the core support plate. Reflector cross-sections are modelled as a function of soluble boron concentration and moderator temperature.

CASMO-3 also generates data that SIMULATE-3 uses to determine the detector reaction rates, based upon detector geometry data input.

2.2.2 TABLES-3 MODEL

The TABLES-3 program generates two-dimensional reactor and cycle specific cross-section tables for SIMULATE-3. Data from the following CASMO-3 card image files are combined into binary cross-section libraries for input into SIMULATE-3.

- HFP Reactor Average Depletion + Branches
  - Fuel Temperature Branches
  - Moderator Temperature Branches
  - Soluble Boron Concentration Branches
  - Control Rod Insertion Branches
  - Shutdown Cooling Branches
- Low Moderator Temperature Depletion
- HFP High Soluble Boron Concentration Depletion
- Bottom Reflector Data
- Radial Reflector Data
- Top Reflector Data

2.2.3 SIMULATE-3 MODEL

The SIMULATE-3 model divides the active fuel region into 16 axial and four radial nodes per assembly. A pseudo-assembly, consisting of reflector material, surrounds the core and is divided into one radial and 16 equal length axial nodes. Axially, the fuel is divided into a single bottom reflector node, 16 nodes for the active fuel region and a single top reflector node.

Additional SIMULATE-3 model input data includes the following:

- Full core assembly serial number map
- Quarter core fuel assembly type map
- Fuel assembly axial zone definition (including reflectors)
- Control rod locations
- Assignment of control rod banks
- In-core instrumentation locations
- Fuel temperature versus power level and burnup correlation (ESCORE)
- Core MW-thermal output at 100% power
- Core pressure, power density and coolant mass flow rate at 100% power conditions
- Coolant inlet temperature versus power level
- Input restart files
- Output restart files

After the cycle base model is set up, the user can specify the percent power level, rod bank positions (inches withdrawn), output and edit options and the type of calculation: depletion, xenon transient or reactivity coefficient calculation (e.g., Isothermal Temperature Coefficient, Inverse Boron Worth, Fuel Temperature Coefficient, etc.).

The FCS reactor is a base loaded facility, meaning operation at or very near rated thermal power throughout the cycle. The lead CEA bank insertion is held to a minimum. Historically, the lead CEA bank at FCS has been inserted less than 5% of the time whenever the reactor is at a

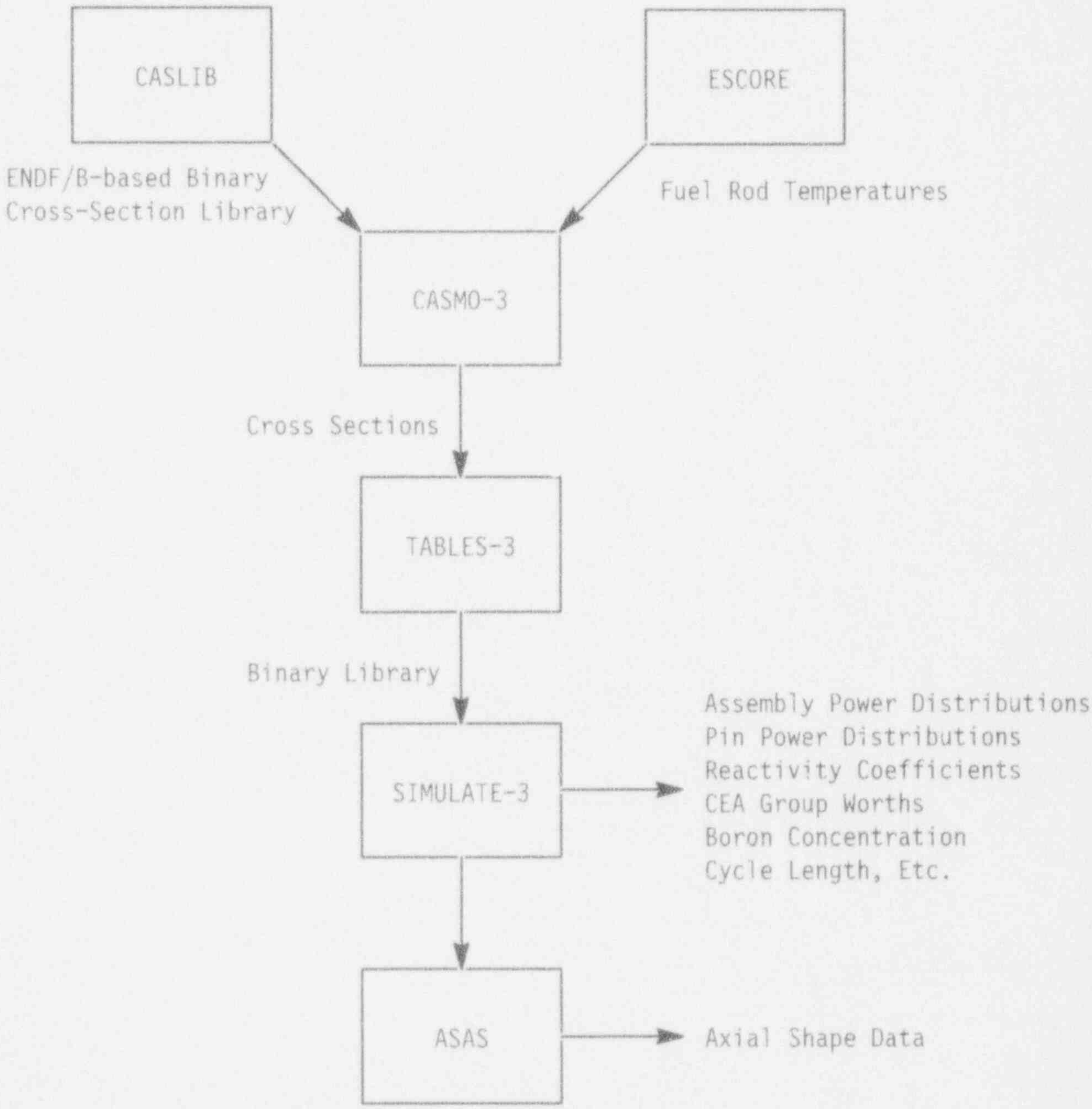
steady power level. Reference 2-8 discusses the impact of operation with a time averaged lead bank insertion of [ ]. Typically, the operating cycle is depleted in time steps of 1 GWD/MTU except for smaller time steps at both the beginning and end of an operating cycle.

2.2.4 AXIAL SHAPE ANALYSIS MODEL

The axial shape analysis consists of axial oscillation power data generated by SIMULATE-3. Axial oscillations are induced in the SIMULATE-3 model by turning off the Doppler feedback and initiating a xenon oscillation. The oscillation is initiated by inserting and withdrawing the lead bank while cycling the reactor power level. This output is fed into the ASAS computer program which processes the pin power distribution data and formats it for use in the calculation of reactor protection system setpoints. ASAS uses the SIMULATE-3 power data along with assembly-wise excore detector view factors to get the excore detector response.

FIGURE 2-1

REACTOR PHYSICS COMPUTER PROGRAM SEQUENCE FLOW CHART



3.0 APPLICATION OF PHYSICS METHODS

The previous section focused on the reactor physics codes and models used by OPPD to explicitly model the FCS reactor. In this section, calculations of the various core parameters used in the safety analysis are described. The primary core parameters considered are the integrated radial and planar radial peaking factors (F_R and F_{XY}), the moderator temperature coefficient of reactivity, the fuel temperature or Doppler coefficient of reactivity, the neutron kinetics parameters, CEA drop data, CEA ejection data, CEA scram reactivity worth, reactivity insertion for the steam line break cooldown, radial peaking data for the asymmetric steam generator event, and axial power distributions. The methods used to develop biases and uncertainties for this document are consistent with previously submitted and NRC-approved OPPD methods.

3.1 RADIAL PEAKING FACTORS

The integrated radial and planar radial peaking factors, F_R and F_{XY}, are calculated using the 3-D SIMULATE-3 model. Values of F_R and F_{XY} for both unrodded and rodded core configurations are obtained directly from the SIMULATE-3 power distribution. The SIMULATE-3 model incorporates a pin power correction to implicitly account for the peaking of the thermal flux in the CEA guide tubes (water holes). The values of F_R and F_{XY} in SIMULATE-3 for unrodded and rodded cores are reported as core peaking edits. SIMULATE-3 calculates F_R based on the axial integration of the planar power distribution. SIMULATE-3 also calculates F_{XY} for each plane to obtain the maximum core F_{XY}. The measurement uncertainties using the CECOR code for the pin peaking factors are given in Reference 3-1.

The reactor physics models are used to calculate the expected values of F_R and F_{XY}. The actual values of F_R and F_{XY} used in the safety analysis are chosen to conservatively bound those anticipated during the core life.

3.2 REACTIVITY COEFFICIENTS

The 3-D SIMULATE-3 model is used to calculate the moderator temperature coefficient (MTC) and the fuel temperature coefficient (FTC). The MTC is defined as the reactivity change associated with a change in moderator inlet

temperature divided by the change in the averaged moderator temperature. The FTC or Doppler coefficient is defined as the reactivity change associated with a uniform change in the fuel temperature divided by the change in the averaged fuel temperature. Both the MTC and FTC calculations first must perform a reference statepoint calculation at base input conditions. The cross-sections and discontinuity factors are then evaluated at the reference conditions except for the perturbation variable which is altered by a user defined amount. A perturbation calculation is then performed, and the coefficients are evaluated based upon the following equation:

$$\frac{\Delta\rho}{\text{Unit change}} = \frac{\Delta\rho}{\Delta CH}$$

where $\Delta\rho = \frac{K_p - K_{ref}}{K_p \cdot K_{ref}}$

and K_{ref} = reference K_{eff}

K_p = perturbed K_{eff}

and $\Delta CH = P_{AVG} - P_{RAVG}$

where P_{AVG} = Volume weighted average of the perturbed parameter

P_{RAVG} = Volume weighted average of the perturbed parameter at reference conditions

The value used for the MTC and FTC perturbations is +5°F.

The reduction in reactivity resulting from an increase in effective fuel temperature is determined by SIMULATE-3. Typically, runs are made at 160, 130, 70 and 40 percent power to get a series of points of fuel temperature coefficient versus fuel temperature while holding moderator temperature and density and nuclide concentrations constant. These points are then fitted to a linear curve to determine the FTC at any fuel temperature.

The physics models are used to calculate the expected values of the MTC and FTC throughout the cycle. The actual values of the MTC and FTC used in the safety analysis are chosen to conservatively bound expected values of these parameters. The measurements of the MTC made during the operation of the reactor include uncertainties to assure that the actual MTC does not exceed the values used in the safety analysis.

3.3 NEUTRON KINETICS PARAMETERS

The data required for point kinetics parameters are generated in SIMULATE-3 by performing spatial integrals over the fueled portion of the 3-D model. Cross sections derived in this manner preserve the core-averaged group-wise reaction rates, leakage rates, and eigenvalue when substituted into the two-group point-reactor diffusion equations. The adjoint flux is used as a weighting function in the definition of the kinetics data.

3.4 DROPPED CEA DATA

The neutronics data unique to the dropped CEA analysis are the values of F_R following the drop of a CEA and the reactivity worth of the dropped CEA. The values of F_R increase due to a large azimuthal tilt caused by the drop of a CEA and occur on the side of the core opposite the dropped CEA. The distortion factor is defined as the ratio of the assembly F_R at a given power level and time in core life containing a dropped CEA to the same assembly F_R without a dropped CEA.

The distortion factor and dropped CEA reactivity worth are calculated using the 3-D SIMULATE-3 model. The 3-D F_R distortion factor is calculated for a specific CEA insertion and power level. The "post-drop" value of F_R, using the 3-D F_R distortion factor, is calculated by multiplying the "pre-drop" value of F_R for the particular CEA insertion and power level by the 3-D F_R distortion factor. The 3-D SIMULATE-3 "post-drop" power distributions are calculated with fuel temperature and moderator temperature feedback. The calculations assume that the core average Axial Shape Index (ASI) is being controlled within the "constant ASI" limits in accordance with the FCS Operating Manual.

3.5 CEA EJECTION DATA

The neutronics data unique to the CEA ejection analysis are values for the pre-ejected and post-ejected radial peaking factors and the reactivity worth of the ejected CEA. The maximum post-ejection radial peaking factor and maximum ejected CEA reactivity worths are calculated for the maximum CEA insertion allowed by the PDIL at HFP and HZP. The neutronics parameters are calculated using HFP and HZP 3-D SIMULATE-3 models. The post-ejection radial

peaking factor, the 3-D peaking factor (F_Q) and the ejected CEA reactivity worth are obtained directly from SIMULATE-3 calculations. SIMULATE-3 post-ejection power distributions are calculated without moderator or fuel temperature feedback.

3.6 CEA REACTIVITY

The CEA reactivity calculations performed in the reload core safety analyses are the calculation of the total reactivity of CEAs inserted into the core during a reactor trip (CEA scram reactivity), the generation of the scram reactivity curves, and the calculation of required shutdown margin.

The CEA scram reactivity worth at HZP is calculated by obtaining the net worth for all CEAs between the HZP PDIL CEA position and the fully inserted position, and subtracting the worth of the highest worth stuck CEA. These calculations are done using the 3-D SIMULATE-3 model. The HZP CEA scram reactivity for the CEA ejection transient is calculated in a similar fashion, except that the worths of the ejected and highest worth stuck CEAs are subtracted from the net worth.

The scram CEA worth at HFP is calculated by obtaining the HFP net worth for all CEAs between the HFP PDIL CEA position and the fully inserted position, subtracting the worth of the highest worth stuck CEA and subtracting the moderator void collapse allowance. The thermal hydraulic axial gradient reduction allowance and the loss of worth between HFP and HZP are also subtracted from the HFP net worth for the scram CEA worth to be used in all transients except the four pump loss of flow event and the main steam line break accident. These are not applied to the four pump loss of flow scram CEA worth because the closest approach to the SAFDL during the four pump loss of flow event occurs prior to significant CEA insertion. These allowances are not applied to the main steam line break (MSLB) incident HFP CEA scram worth because the HFP MSLB reactivity insertion curves implicitly account for these effects.

The axial thermal hydraulic reactivity effects for HFP to HZP transients are accounted for by measuring the change in reactivity resulting from collapsing a 3-D distributed moderator temperature to a 3-D flat moderator temperature profile. This is done by setting the power in a SIMULATE-3 case to HZP from

a HFP input file and allowing only fuel temperature to feedback. In the next case, the core average moderator temperature from the HFP case is input to the HZP case as the inlet temperature and the fuel temperature are frozen, only allowing the moderator density and temperature to feedback. The bounding values at Beginning-of-Cycle (BOC) and End-of-Cycle (EOC) are used for calculation of scram worths for transient analyses.

The generation of the scram reactivity curves uses the methodology discussed in Reference 3-2.

The calculation of the required shutdown margin is only performed at HZP since the shutdown margin at power is controlled by the PDIL. The available HZP shutdown margin is equivalent to the HZP CEA scram reactivity.

3.7 CEA WITHDRAWAL DATA

The reactor core physics data unique to the CEA withdrawal analysis is the maximum differential CEA worth. This is the maximum amount of reactivity at any time in core life that can be added to the core per inch of CEA motion.
[

] This calculation produces a more conservative estimate of the expected maximum differential worth than previous methods.

This maximum differential worth is then combined with the maximum CEA withdrawal rate of 46 inches/minute to arrive at the maximum reactivity insertion rate.

3.8 REACTIVITY INSERTION FOR MAIN STEAM LINE BREAK COOLDOWN

The reactor core physics data unique to the main steam line break accident analysis is the reactivity insertion due to the cooldown of the moderator. There are two sources of this reactivity insertion. The first is the positive reactivity insertion due to the increasing density of the moderator as the cooldown progresses. The second is the reactivity insertion due to the FTC as the effective fuel temperature changes.

Reactivity insertions due to increases in the moderator density and FTC changes are both calculated using a full core SIMULATE-3 model. The axial leakage or buckling is adjusted such that the MTC calculated by the SIMULATE-3 model corresponds to the most negative Technical Specification limit. The reactivity insertion calculations are performed with all CEAs except the most reactive CEA inserted in the core.

The moderator density reactivity insertion curve for the HZP main steam line break case is calculated by successively lowering the inlet temperature of the SIMULATE-3 model from 532°F and allowing only moderator temperature feedback in the model. The calculations typically result in a curve of reactivity insertion versus moderator temperature from a HZP temperature of 532°F to 212°F.

The Doppler reactivity insertion for the HZP case bounds the value of the reactivity insertion calculated by SIMULATE-3 for the FTC. The fuel temperature feedback in the model allows the production of a curve of Doppler reactivity as a function of fuel temperature. All zero power calculations are performed assuming there is no decay heat and no credit is taken for local voiding in the region of the stuck CEA.

The moderator density reactivity insertion curve for the full power case is calculated by decreasing the power level and core average average coolant temperature from full power to the HZP inlet temperature and then successively lowering the inlet temperature as in the HZP case. Only moderator temperature feedback is used in the SIMULATE-3 model.

Since the moderator reactivity insertion curve corresponds to an MTC that is at the Technical Specification limit, no additional uncertainty is added to this curve.

3.9 ASYMMETRIC STEAM GENERATOR EVENT DATA

The reactor core physics data unique to the asymmetric steam generator event [ ] For the range of temperatures considered, the intra-assembly peaking does not vary as the inlet temperature is changed. [

]

[

]

3.10 AXIAL SHAPE ANALYSIS CALCULATIONS

The ASAS data input to the setpoint analysis includes axial power distribution, power-to-fuel design limits, axial shape, rod bank insertion, excore detector response, F_Z, F_Q and the location of the F_Z and F_Q peaks. Uncertainties for these parameters are applied in the calculation of setpoints consistent with Reference 3-3, except for the F_Q uncertainty which is applied consistent to the methods described in Reference 3-4.

A description of the SIMULATE-3 code is presented in section 2.1.5. The use of a single physics model (i.e., SIMULATE-3) for the generation of the power-to-fuel design limit simplifies the analysis and results in a more detailed modelling of axial transients.

4.0 BENCHMARK OF CASMO-3/SIMULATE-3 MODELS

OPPD has performed extensive benchmarking of the CASMO-3/SIMULATE-3 neutronics models used in the reload core analyses. The measured data base consists of data from the five most recent low power physics startup tests (Cycles 11, 12, 13, 14 and 15) and from normal operations during the four most recently completed operating cycles (Cycles 11, 12, 13, 14), as well as from the current operating cycle (Cycle 15) at FCS.

For the critical boron concentrations, isothermal temperature coefficients and control rod worths, 95/95 probability/confidence limits were calculated. The probability/confidence limits were applied to the indicated CASMO-3/SIMULATE-3 results such that there is a 95 percent probability with a 95 percent confidence that the calculated values will conservatively bound the measured (i.e., "true") values. For the assembly pin peaking factors, biases for the SIMULATE-3 predictions were established to duplicate the MC predictions that are consistent with previously NRC-approved OPPD methods.

The results of the previous OPPD verification efforts were reported in References 4-1 through 4-4.

4.1 CRITICAL BORON CONCENTRATION

SIMULATE-3 Critical Boron Concentration (CBC) predictions were compared to low power physics startup test measurements (zero power) and full power operating measurements. The zero power startup test measurements were taken during the five most recent operating cycle startups (Cycles 11 through 15) under well controlled conditions without significant thermal and xenon feedbacks. Both unrodded and rodded configurations are represented in the zero power startup test measurement data base. The full power measurement data were collected during the four most recently completed operating cycles (Cycles 11 through 14) and part of the current operating cycle (Cycle 15).

Table 4-1 summarizes the comparison of the zero power SIMULATE-3 CBC predictions with the low power physics CBC measurements. Zero power startup predictions from ROCS are also included to provide an additional comparison. The BOC, HZP SIMULATE-3 CBC predictions, both unrodded and rodded, compare favorably with the corresponding zero power startup CBC measurements.

Figures 4-1 through 4-5 provide comparisons of the HFP SIMULATE-3 CBC predictions with the CBC at-power measurements. The CBC measured data were

adjusted for control rod insertions and deviations from full power, equilibrium conditions. The HFP SIMULATE-3 CBC predictions show excellent agreement with the CBC rundown measurements.

In an effort to demonstrate OPPD's ability to predict boron concentrations for shutdown margin requirements at cold conditions (68°F to 532°F) with CASMO-3/SIMULATE-3, the Babcock & Wilcox (B&W) cold critical experiments from References 4-5 and 4-6 were modelled by OPPD with CASMO-3/SIMULATE-3. OPPD's SIMULATE-3 prediction of K_{eff} is consistent between the Fort Calhoun Station model at HZP and the results from the B&W reports. These results are also consistent with the work performed by Studsvik (Reference 4-7). Thus, the conclusion from these comparisons is that OPPD can adequately and conservatively determine the minimum boron concentrations for shutdown margin using SIMULATE-3.

4.2 ISOTHERMAL TEMPERATURE COEFFICIENT

SIMULATE-3 Isothermal Temperature Coefficient (ITC) predictions were compared to low power physics startup test measurements (zero power) and at-power operating measurements. The zero power startup test measurements were taken during the five most recent operating cycle startups (Cycles 11 through 15). The at-power measurement data were collected during the four most recently completed operating cycles (Cycles 11 through 14) and part of the current operating cycle (Cycle 15).

Table 4-2 summarizes the comparison of the zero power SIMULATE-3 ITC predictions with the low power physics ITC measurements. Zero power startup predictions from ROCS are also included to provide an additional comparison. The BOC, HZP SIMULATE-3 ITC predictions compare favorably with the zero power startup ITC measurements.

Table 4-3 provides a comparison of the at-power SIMULATE-3 ITC predictions with the ITC measurements. The at-power measurements were taken during each of the at-power MTC test programs required by Technical Specifications. At-power ITC predictions from ROCS are also included as an additional comparison. The at-power SIMULATE-3 ITC predictions show good agreement with the ITC at-power measurements.

4.3 POWER COEFFICIENT

SIMULATE-3 Power Coefficient (PC) predictions were compared to at-power measurements collected during the four most recent operating cycles (Cycles 11 through 14) and a portion of the current operating cycle (Cycle 15).

Table 4-4 summarizes the comparison of the at-power SIMULATE-3 PC predictions with the at-power PC measurements. The at-power PC measurements were taken during each of the at-power MTC test programs required by Technical Specifications. At-power PC predictions from ROCS are also included to provide an additional comparison. The at-power SIMULATE-3 PC predictions compare favorably with the at-power PC measured data.

4.4 CONTROL ROD WORTH

SIMULATE-3 Control Element Assembly (CEA) group worth predictions were compared to measurements from low power physics tests since Cycle 11. Tables 4-5 through 4-9 summarize the comparison of the low power physics startup test SIMULATE-3 predictions with the zero power startup test measurements. Low power physics CEA worth predictions from ROCS are also included as an additional comparison. The low power physics startup test SIMULATE-3 CEA Worth predictions show excellent agreement with the zero power startup test CEA worth measurements.

4.5 ASSEMBLY RELATIVE POWER DISTRIBUTIONS

SIMULATE-3 predictions for assembly power distributions, both radially and axially, were compared to core follow data starting with Cycle 11. Figures 4-6 through 4-18 summarize the comparison between the axially integrated CECOR assembly relative power density (RPD) measurements with the SIMULATE-3 assembly RPD predictions at nominal power levels for BOC, Middle-Of-Cycle (MOC), and EOC conditions. These figures show comparisons for only the assemblies containing in-core instruments. Limiting the comparison to the assembly locations containing in-core instruments provides the most direct comparison to actual assembly powers and limits the amount of ROCS-based information. By doing so, the comparison between the SIMULATE-3 predictions and CECOR measurements in only in-core instrumented locations represents a more accurate benchmark of the SIMULATE-3 FCS model. The comparisons show

good agreement between the CECOR RPD measurements and SIMULATE-3 RPD predictions.

Figures 4-19 through 4-31 summarize the comparison between the normalized CECOR axial power distribution measurements with the SIMULATE-3 axial power distribution predictions at nominal power levels for BOC, MOC and EOC conditions. The comparisons show consistent agreement between the CECOR axial power measurements and SIMULATE-3 axial power predictions.

4.6 ASSEMBLY PIN PEAKING FACTORS

SIMULATE-3 predictions for integrated radial peaking factors (F_R), planar radial peaking factors (F_{XY}) and 3-D peaking factors (F_Q) were compared to CECOR pin peaking factors from core follow data starting with Cycle 11. Figures 4-32 through 4-36 summarize the comparison between CECOR maximum F_R measurements and SIMULATE-3 peak F_R predictions. Figures 4-37 through 4-41 summarize the comparison between CECOR maximum F_{XY} measurements and SIMULATE-3 peak F_{XY} predictions. Figures 4-42 through 4-46 summarize the comparison between CECOR maximum F_Q measurements and SIMULATE-3 peak F_Q predictions. The SIMULATE-3 predictions were based upon full power operations, while the CECOR measurements include nominal full power data as well as off-nominal power level pin peaking values. All CECOR measurement data included a component for tilt. Peaking factor spikes from the CECOR measured data occur during power reductions and are considered normal. The comparisons show good agreement between the CECOR maximum pin power measurements and SIMULATE-3 maximum pin power predictions.

4.7 SUMMARY

OPPD continues to maintain an ongoing neutronics methodology verification program. Verification of program segments include information consisting of zero power startup physics testing predictions, reactor at-power testing analyses and core follow efforts. The results of this verification program for previous cycles demonstrate the ability of OPPD personnel to use the CASMO-3/SIMULATE-3 neutronics methods described in this document.

5.0 CONCLUSIONS

Omaha Public Power District (OPPD) has performed extensive benchmarking to incorporate the CASMO-3/SIMULATE-3 computer code system into neutronics design methods for Fort Calhoun Station Unit No. 1. This effort consisted of comparisons of core physics predictions to measurements from previous Fort Calhoun Station operating cycles. The results of this benchmark program for previous operating cycles demonstrate the ability of OPPD to use the CASMO-3/SIMULATE-3 neutronics methods described in this document.

OPPD has continually demonstrated its ability to perform core neutronics calculations for reload core analysis as documented in References 5-1 through 5-4. Based upon the analyses and results contained in this document, as well as OPPD's current ability to perform neutronics reload analyses, OPPD concludes that the methodology using CASMO-3/SIMULATE-3 computer code system applies to all steady-state reactor physics calculations. The accuracy of this methodology is sufficient for use in licensing applications, reload design depletion analyses, reactor physics safety analyses, startup physics predictions, core physics databooks, and reactor protection system and monitoring system setpoint upgrades.

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1-2 M. Edenius, C. Gragg, "CASLIB User's Manual, Version 1.3," STUDSVIK/NFA - 89/13, Studsvik Energiteknik AB, Sweden, November 1989.

1-3 K. S. Smith, J. A. Umbarger, D. M. Ver Planck, "TABLES-3, Library Preparation Code for SIMULATE-3, User's Manual, Version 3.0," STUDSVIK/SOA - 89/05, Studsvik of America, Inc., November 1989.

1-4 K. S. Smith, J. A. Umbarger, D. M. Ver Planck, "SIMULATE-3, Advanced Three-Dimensional Two-Group Reactor Analysis Code, User's Manual, Version 3.0," STUDSVIK/SOA - 89/03, Studsvik of America, Inc., November 1989.

1-5 "CASMO-3G Validation", YAEC-1363-A, Yankee Atomic Electric Company, April 1988.

1-6 "SIMULATE-3 Validation and Verification", YAEC-1659-A, Yankee Atomic Electric Company, September 1988.

Section 2.0 References

2-1 "ESCORE: The EPRI Steady-State Core Reload Evaluator Code; General Description," EPRI NP-5100-A, May 1990.

2-2 Letter, USNRC to C. R. Lehmann (PP&L), "Acceptance for Referencing of Licensing Topical Report EPRI-NP-5100, ESCORE - The EPRI Steady-State Core Reload Evaluation Code: General Description," May 23, 1990.

2-3 M. Edenius, C. Gragg, "CASLIB User's Manual, Version 1.3," STUDSVIK/NFA - 89/13, Studsvik Energiteknik AB, Sweden, November 1989.

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2-5 K. S. Smith, J. A. Umbarger, D. M. Ver Planck, "TABLES-3, Library Preparation Code for SIMULATE-3, User's Manual, Version 3.0," STUDSVIK/SOA - 89/05, Studsvik of America, Inc., November 1989.

2-6 K. S. Smith, J. A. Umbarger, D. M. Ver Planck, "SIMULATE-3, Advanced Three-Dimensional Two-Group Reactor Analysis Code, User's Manual, Version 3.0," STUDSVIK/SOA - 89/03, Studsvik of America, Inc., November 1989.

2-7 Theoretical Manual and User's Manual for Axial Shape Analyzer for SIMULATE-3 (ASAS), April 19, 1994.

2-8 CENPD-199-P, Revision 1-P-A, "CE Setpoint Methodology," January 1986.

Section 3.0 References

3-1 CENPD-153, Revision 1-P-A, "INCA/CECOR Power Peaking Uncertainty," May, 1980.

3-2 CENPD-199-P, Revision 1-P-A, "CE Setpoint Methodology," January 1986.

3-3 "Statistical Combination of Uncertainties," CEN-257(0)-P-A, Parts 1, 2, and 3, November 1983, including Supplement 1-P, August 1985.

3-4 Maine Yankee SER on YAEC-1110, "Maine Yankee Reactor Protection System Setpoint Methodology," dated May 27, 1977.

Section 4.0 References

4-1 CEN-242-(0)-P, OPPD Responses to NRC Questions on Fort Calhoun Cycle 8, February 18, 1983.

4-2 "Reload Core Analysis Methodology, Neutronics Design Methods and Verification," OPPD-NA-8302-P-A, Rev. 01.

4-3 "Reload Core Analysis Methodology, Neutronics Design Methods and Verification," OPPD-NA-8302-P-A, Rev. 02.

4-4 "Reload Core Analysis Methodology, Neutronics Design Methods and Verification," OPPD-NA-8302-P-A, Rev. 03.

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4-7 "SIMULATE-3 Pin Power Reconstruction: Benchmarking Against the B&W Critical Experiments," Kord S. Smith (Studsvik of America, Inc.), Transactions of the American Nuclear Society, Volume 56, TANSAO 56 1-628 (1988), June 12-16, 1988.

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5-3 "Reload Core Analysis Methodology, Neutronics Design Methods and Verification," OPPD-NA-8302-P-A, Rev. 02.

5-4 "Reload Core Analysis Methodology, Neutronics Design Methods and Verification," OPPD-NA-8302-P-A, Rev. 03.

TABLE 4-1

COMPARISON OF ZERO POWER CRITICAL BORON CONCENTRATIONS
(BOC, NO XENON)
(ppm)

Cycle	CEA Position	Measured	3-D ROCS-DIT	3-D SIMULATE-3	(SIMULATE-3 - Measured)
11	ARO	1502		1501	-1
11	Group-4 ARI	1421		1424	+3
11	Group-2 ARI	1399		1398	-1
11	Group-A ARI	1303		1299	-4
12	ARO	1507		1514	+7
12	Group-4 ARI	1445		1446	+1
12	Group-2 ARI	1422		1427	+5
12	Group-A ARI	1336		1331	-5
13	ARO	1568		1572	+4
13	Group-4 ARI	1524		1524	0
13	Group-1 ARI	1483		1478	-5
13	Group-A ARI	1385		1385	0
14	ARO	1182		1179	-3
14	Group-4 ARI	1113		1115	+2
14	Group-B ARI	1009		1001	-8
15	ARO	1411		1392	-19
15	Group-4 ARI	1366		1347	-19
15	Group-B ARI	1227		1202	-25

TABLE 4-2

COMPARISON OF LOW POWER PHYSICS ISOTHERMAL TEMPERATURE COEFFICIENTS
($\times 10^{-4}\ \Delta\rho/{}^{\circ}F$)

Cycle	Measured	3-D ROCS-DIT	3-D SIMULATE-3
11	0.20		0.20
12	0.24		0.23
13	0.32		0.30
14	-0.09		-0.02
15	0.10		0.14

TABLE 4-3

COMPARISON OF AT-POWER CALCULATED AND MEASURED
ISOTHERMAL TEMPERATURE COEFFICIENTS
(x 10^{-4} $\Delta\rho$/°F)

BOC

Cycle	Percent Power	CBC (ppm)	Measured	3D ROCS-DIT	3D SIMULATE-3
11	93	1073	-0.43		-0.50
12	93	1050	-0.53		-0.55
13	94	1113	-0.51		-0.48
14	90	768	-0.87		-0.81
15	95	948	-0.81		-0.69

EOC

Cycle	Percent Power	CBC (ppm)	Measured	3D ROCS-DIT	3D SIMULATE-3
11	95	301	-1.62		-1.72
12	95	309	-1.79		-1.74
13	95	325	-1.69		-1.65
14	94	319	-1.76		-1.78

NOTE: Full Rated Power = 1500 MWt

TABLE 4-4

COMPARISON OF CALCULATED AND MEASURED POWER COEFFICIENTS
(x 10^{-4} $\Delta\rho$/% Power)

Cycle	Burnup (MWD/MTU)	Percent Power	CBC (ppm)	Measured	3D ROCS-DIT	3D SIMULATE-3
11	433	93	1073	-0.95		-0.98
11	9765	95	301	-1.52		-1.38
12	425	93	1050	-1.42		-1.00
12	9691	95	309	-1.63		-1.41
13	373	94	1113	-1.26		-1.00
13	10694	95	325	-1.51		-1.41
14	355	90	768	-1.55		-1.16
14	10559	93	319	-1.77		-1.52
15	340	95	948	-1.57		-1.02

NOTE: Full Rated Power = 1500 MWt

TABLE 4-5

COMPARISON OF CYCLE 11 BOC, HZP CEA WORTHS
($\%\Delta\rho$)

Group	Measured	3-D ROCS-DIT	3-D SIMULATE-3
A	1.96		1.99
B	1.28		1.36
4	0.76		0.76
3	0.46		0.50
2	0.97		0.99
1	0.58		0.65
Total	6.01		6.25

TABLE 4-6

COMPARISON OF CYCLE 12 BOC, HZP CEA WORTHS
(%$\Delta\rho$)

Group	Measured	3-D ROCS-DIT	3-D SIMULATE-3
A	1.92		1.80
B	1.52		1.45
4	0.61		0.66
3	0.56		0.65
2	0.80		0.83
1	0.63		0.71
Total	6.04		6.10

TABLE 4-7

COMPARISON OF CYCLE 13 BOC, HZP CEA WORTHS
($\%\Delta\rho$)

Group	Measured	3-D ROCS-DIT	3-D SIMULATE-3
A	1.80		1.81
B	1.56		1.53
4	0.45		0.46
3	0.73		0.79
2	1.12		1.08
1	0.81		0.89
Total	6.47		6.56

TABLE 4-9

COMPARISON OF CYCLE 15 BOC, HZP CEA WORTHS
($\%\Delta\rho$)

Group	Measured	3-D ROCS-DIT	3-D SIMULATE-3
A	1.51		1.55
B	1.60		1.76
4+3	0.99		1.08
2+1	1.57		1.65
Total	5.67		6.04

Figure 4-1

Cycle 11 Critical Boron Concentration vs Burnup

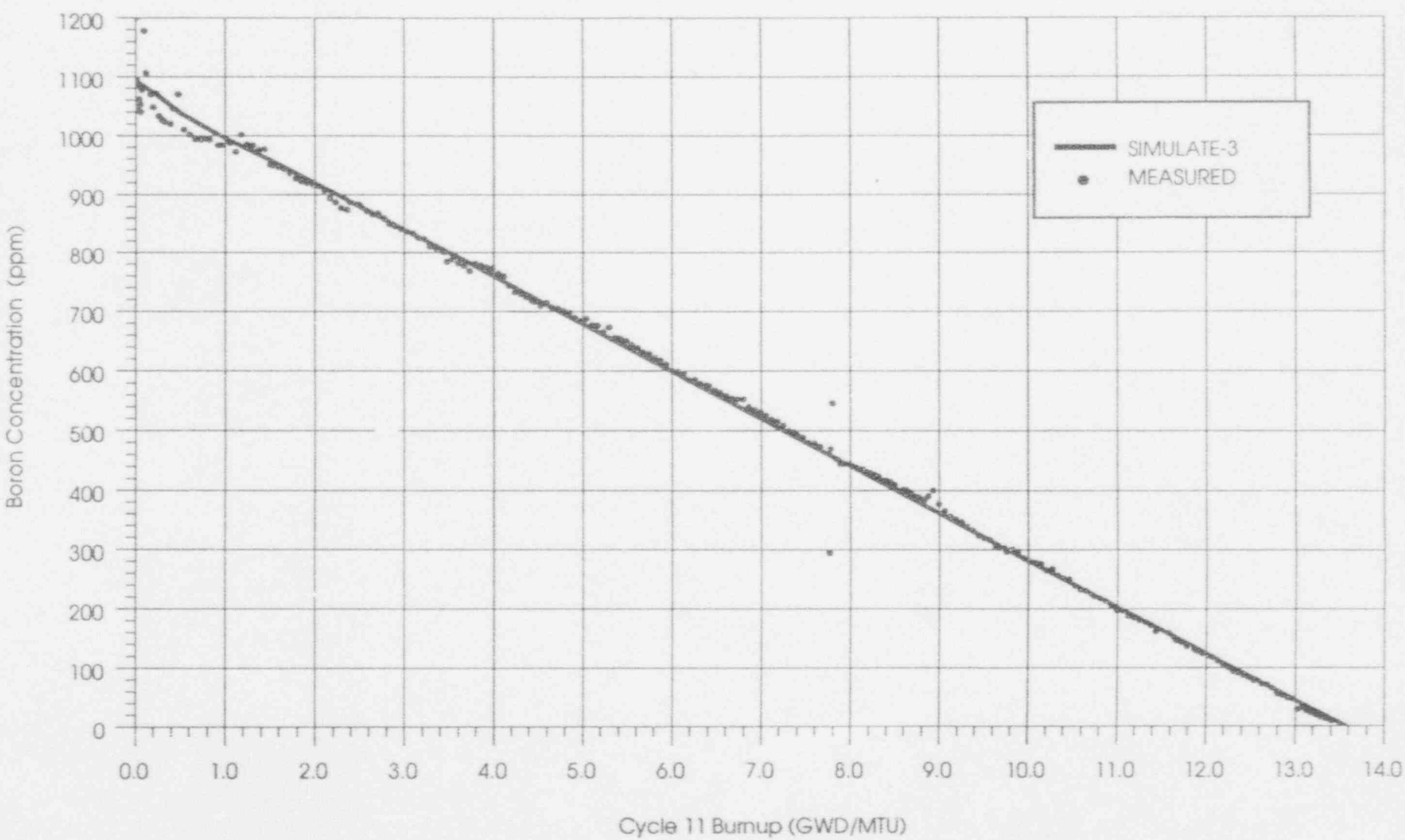


Figure 4-2

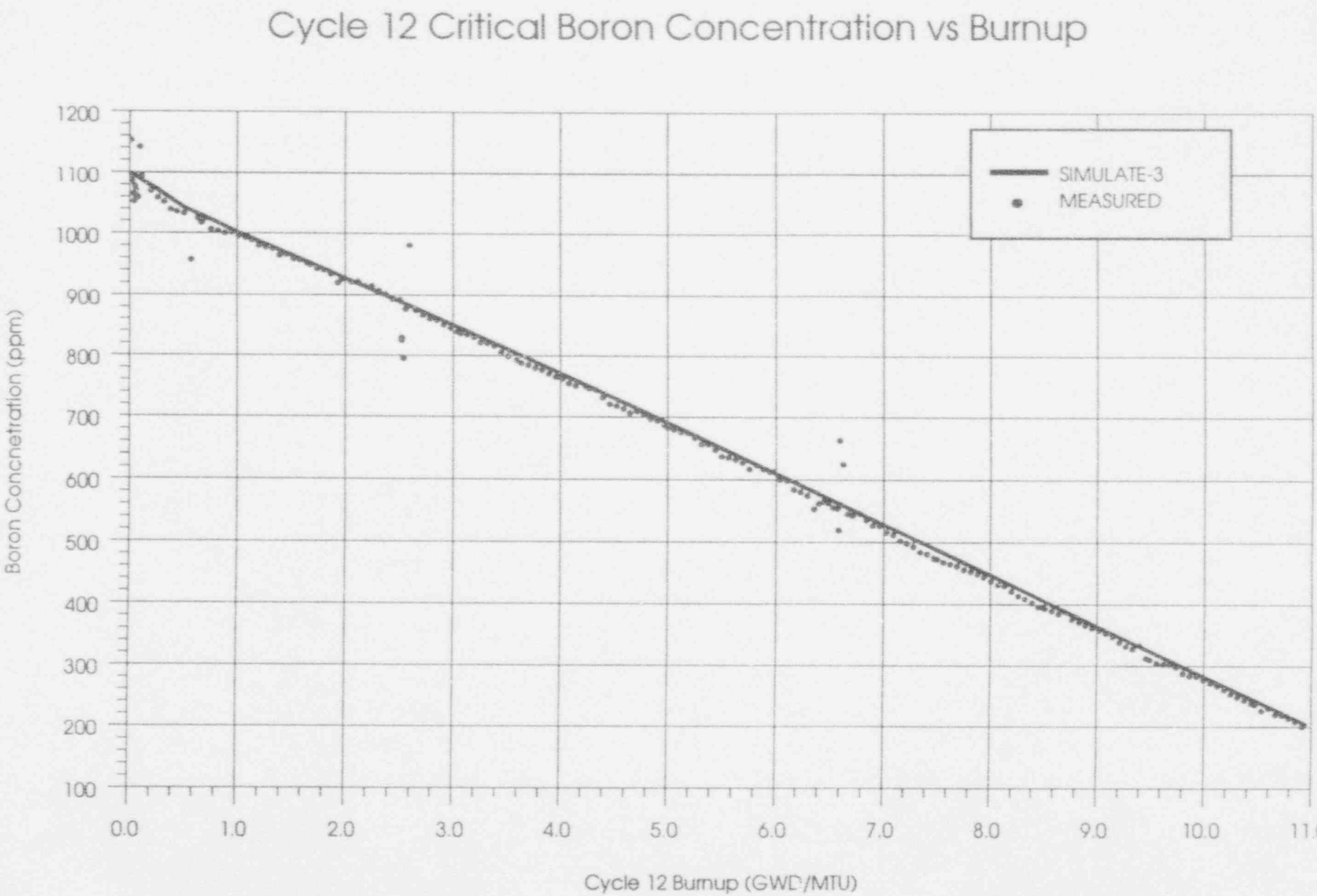

Cycle 12 Critical Boron Concentration vs Burnup
SIMULATE-3
MEASURED
Boron Concnetration (ppm)
1200
1100
1000
900
800
700
600
500
400
300
200
100
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
Cycle 12 Burnup (GWD/MTU)

Figure 4-3

Cycle 13 Critical Boron Concentration vs Burnup

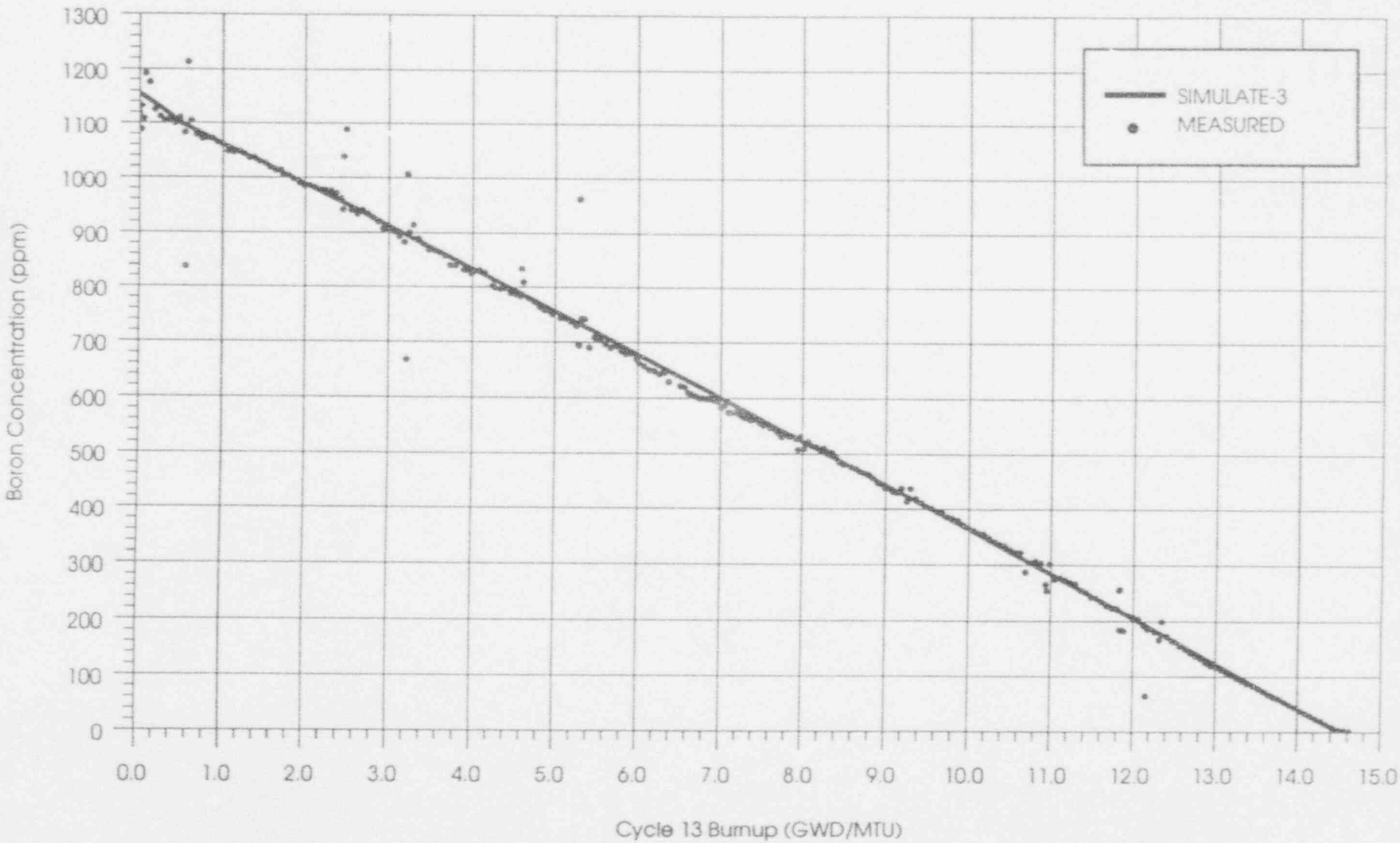
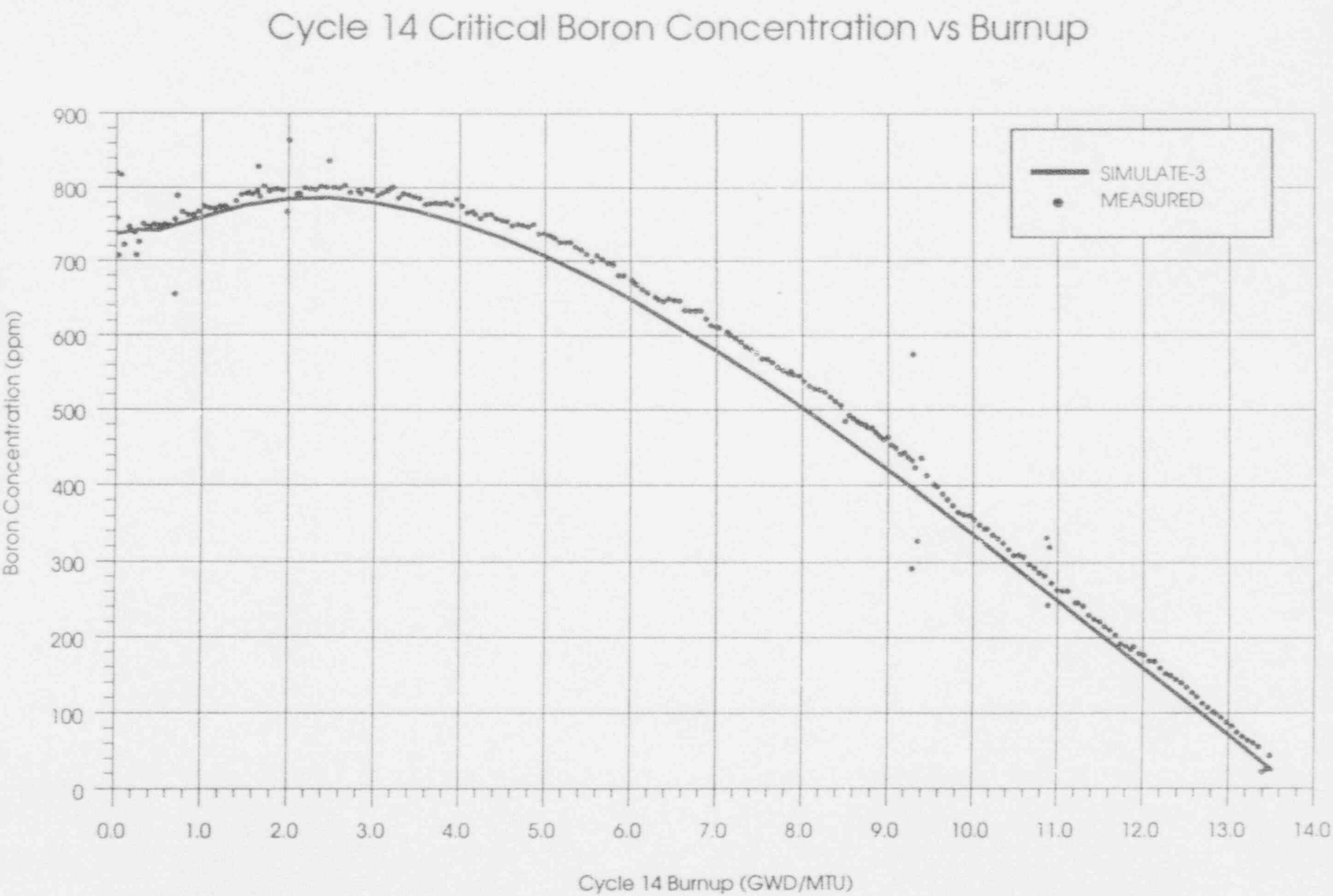


Figure 4-4



Cycle 14 Critical Boron Concentration vs Burnup
SIMULATE-3
MEASURED
Boron Concentration (ppm)
900
800
700
600
500
400
300
200
100
0
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
11.0
12.0
13.0
14.0
Cycle 14 Burnup (GWD/MTU)

Figure 4-5

Cycle 15 Boron Concentration vs Exposure (GWD/MTU)

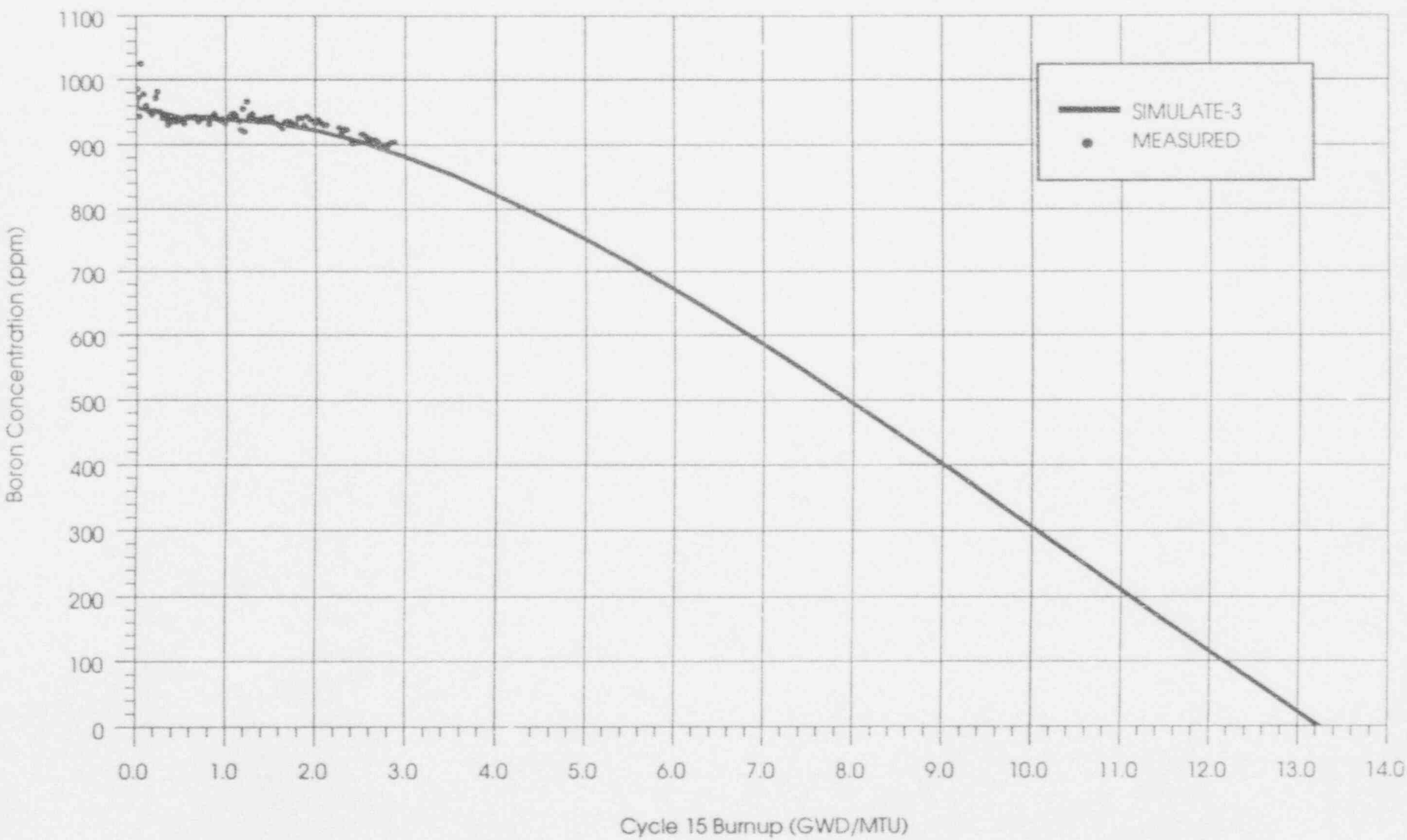
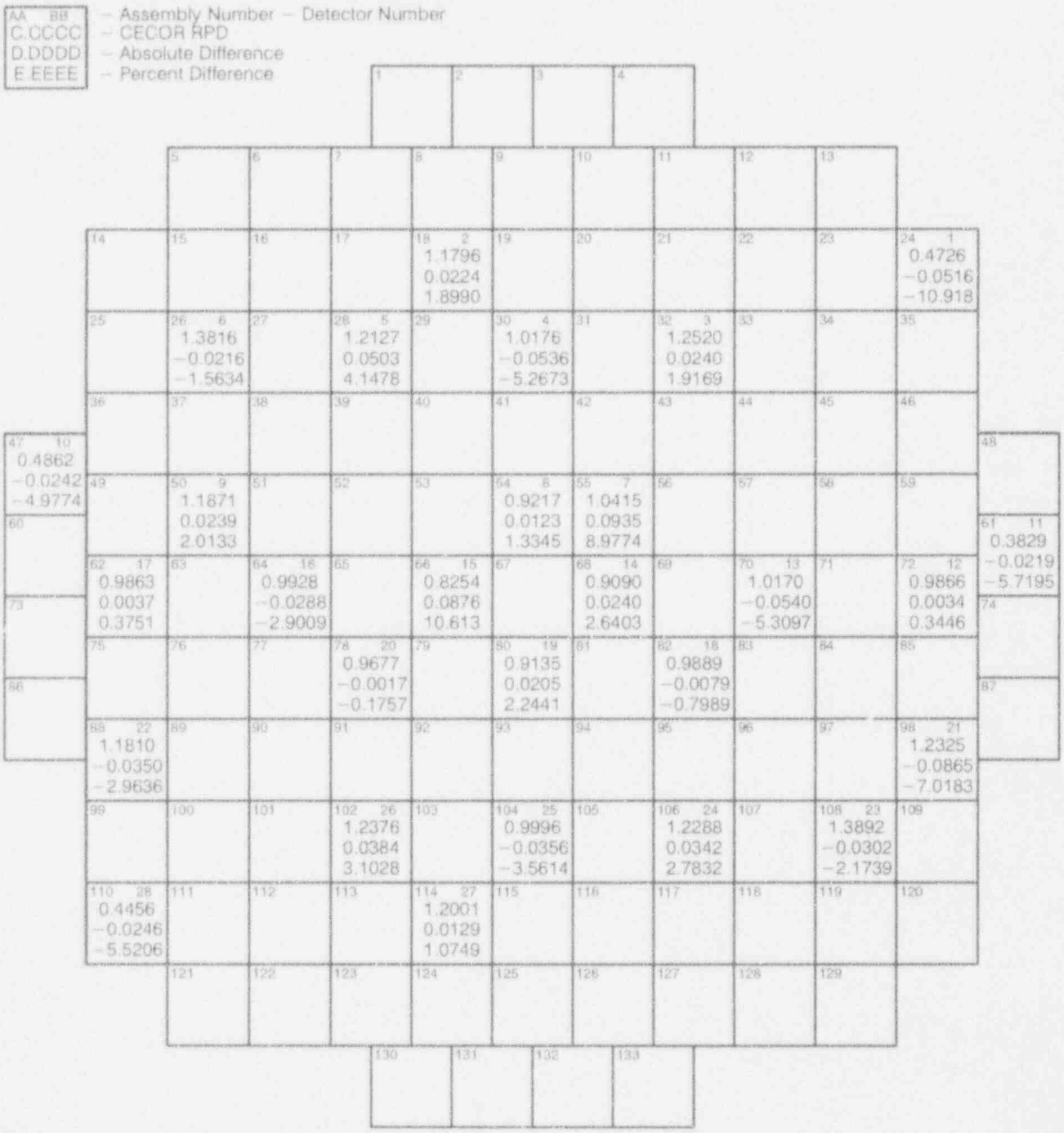


Figure 4-6

SIMULATE – 3/CECOR Radial Power Comparison

Cycle 11 at 1,094 MWD/MTU, 99.4% Power, 973 ppm
Axially Integrated



SIMULATE Case Values:
- Average = 1.0004
- RMS Error = 4.0948
- Maximum Value = 1.3600 at Detector 6
- Minimum Value = 0.3610 at Detector 11

CECOR Case Values:
- Average = 1.0013
- Standard Deviation = 0.2718
- Maximum Value = 1.3892 at Detector 23
- Minimum Value = 0.3829 at Detector 11

Absolute Differences:
- Average = –0.0009
- Standard Deviation = 0.0417
- Maximum Value = 0.0935 at Detector 7
- Minimum Value = –0.0865 at Detector 21

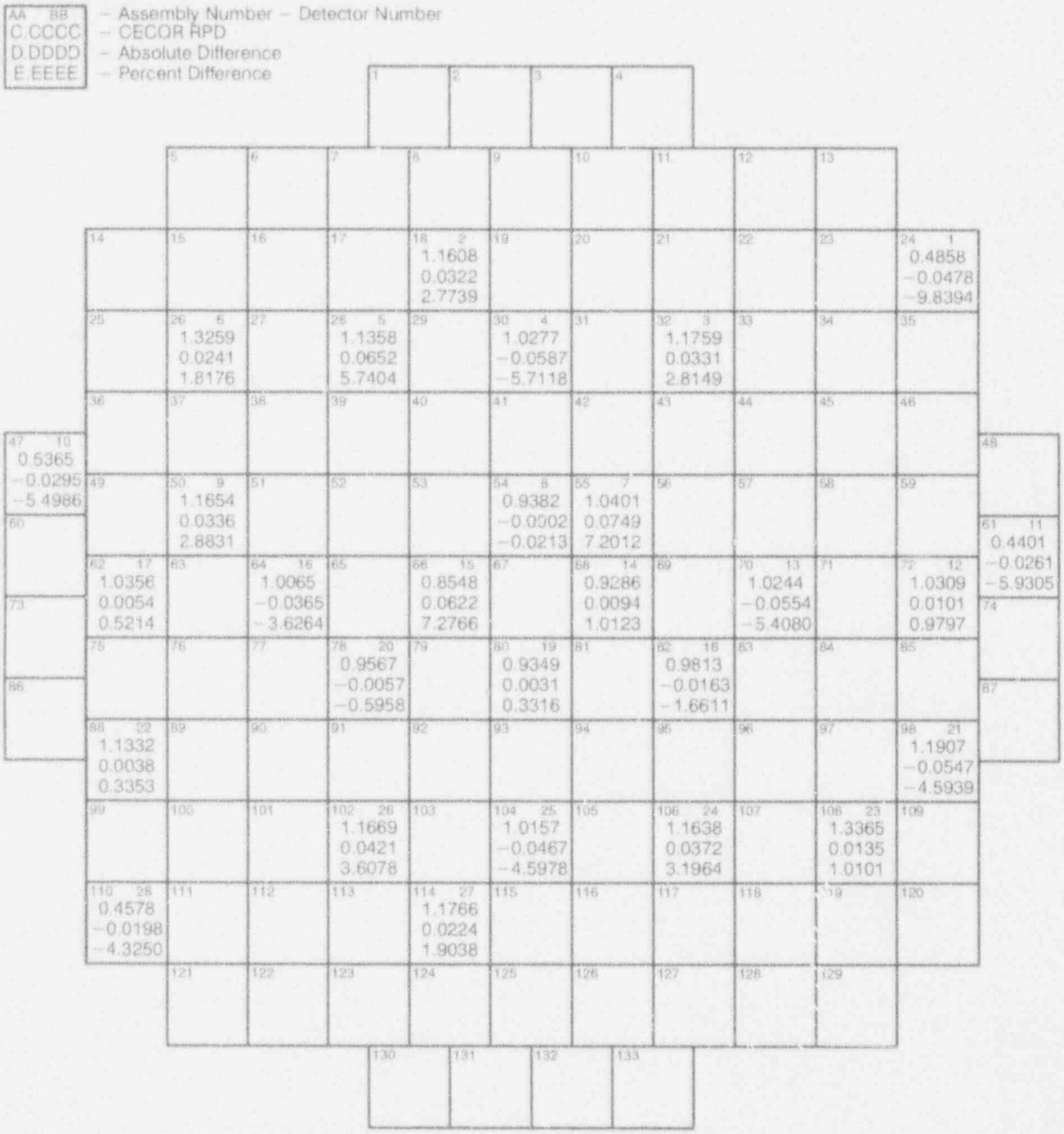
Percentage Differences:
- Average = –0.5501
- Standard Deviation = 4.6982
- Maximum Value = 10.6130 at Detector 15
- Minimum Value = –10.9183 at Detector 1

Figure 4-7

SIMULATE–3/CECOR Radial Power Comparison

Cycle 11 at 6,990 MWD/MTU, 99.5% Power, 532 ppm

Axially Integrated



SIMULATE Case Values:
- Average = 0.9965
- RMS Error = 3.7418
- Maximum Value = 1.3500 at Detector 6
- Minimum Value = 0.4140 at Detector 11

CECOR Case Values:
- Average = 0.9938
- Standard Deviation = 0.2426
- Maximum Value = 1.3365 at Detector 23
- Minimum Value = 0.4401 at Detector 11

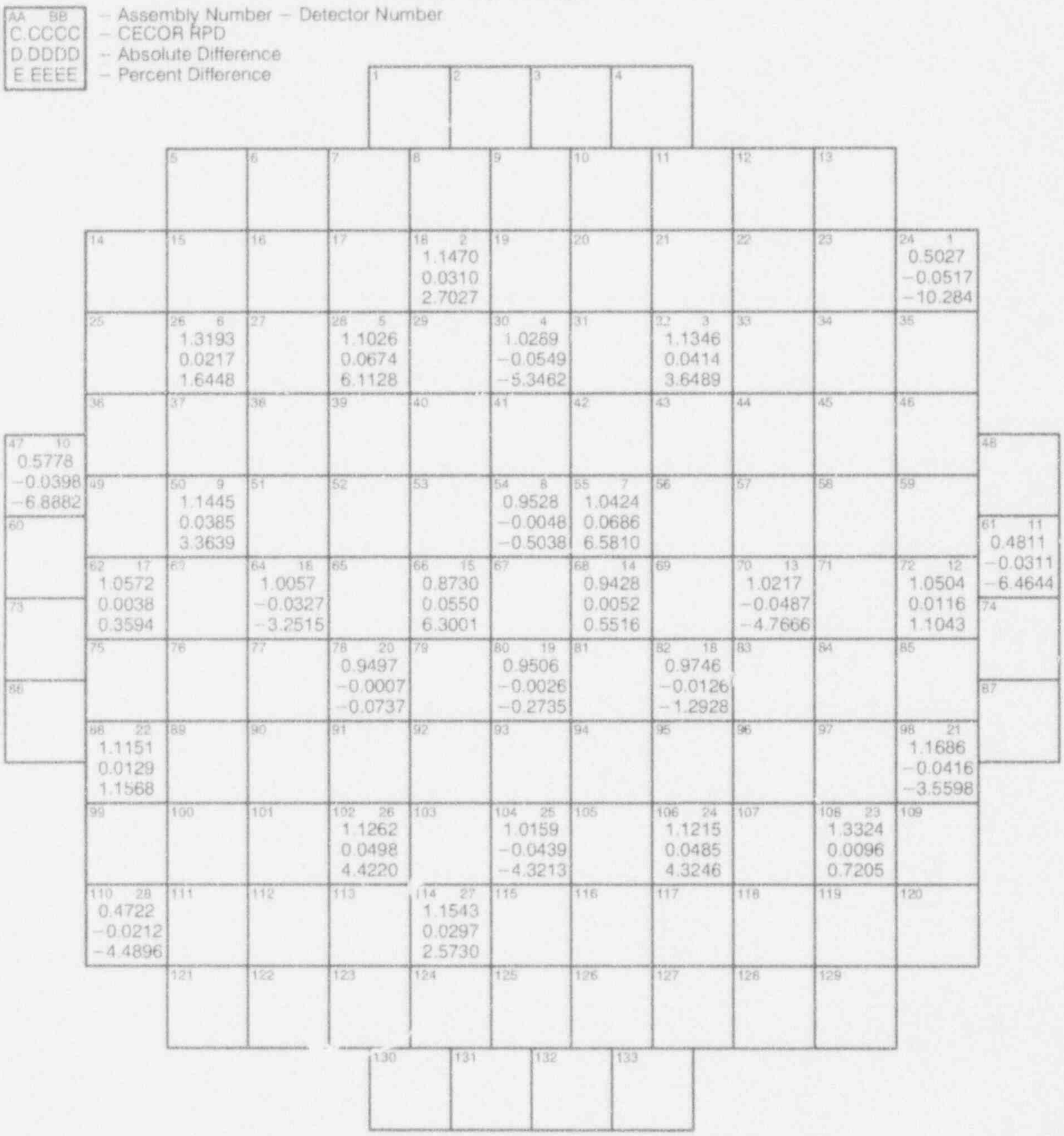
Absolute Differences:
- Average = 0.0027
- Standard Deviation = 0.0380
- Maximum Value = 0.0749 at Detector 7
- Minimum Value = −0.0587 at Detector 4

Percentage Differences:
- Average = −0.3001
- Standard Deviation = 4.2780
- Maximum Value = 7.2766 at Detector 15
- Minimum Value = −9.8394 at Detector 1

Figure 4-8

SIMULATE–3/CECOR Radial Power Comparison

Cycle 11 at 11,088 MWD/MTU, 99.3% Power, 189 ppm
Axially Integrated



SIMULATE Case Values:
- Average = 0.9954
- RMS Error = 3.7334
- Maximum Value = 1.3420 at Detector 23
- Minimum Value = 0.4500 at Detector 11

CECOR Case Values:
- Average = 0.9916
- Standard Deviation = 0.2267
- Maximum Value = 1.3324 at Detector 23
- Minimum Value = 0.4722 at Detector 28

Absolute Differences:
- Average = 0.0039
- Standard Deviation = 0.0378
- Maximum Value = 0.0686 at Detector 7
- Minimum Value = −0.0549 at Detector 4

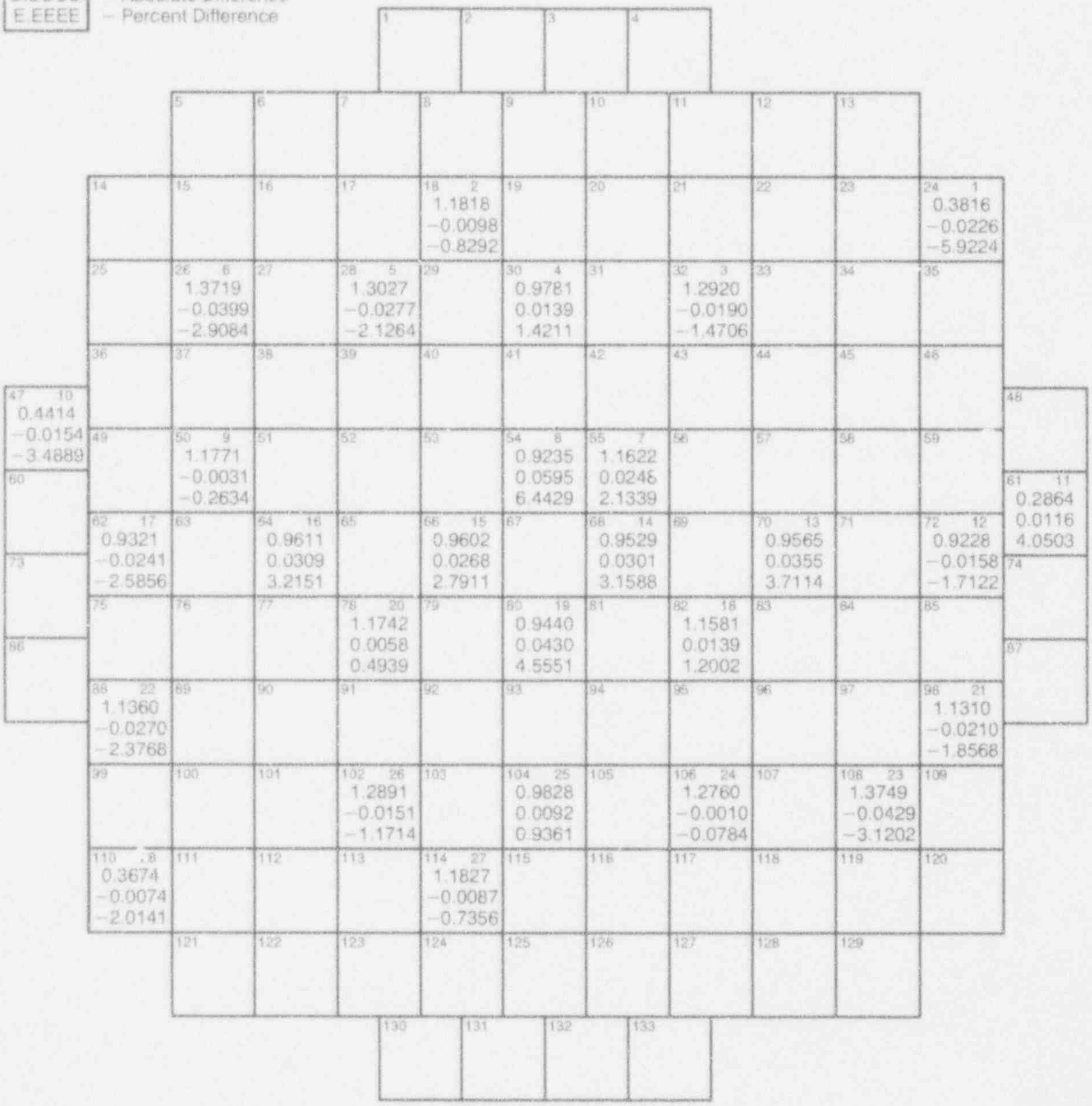
Percentage Differences:
- Average = −0.2125
- Standard Deviation = 4.3568
- Maximum Value = 6.5810 at Detector 7
- Minimum Value = −10.2845 at Detector 1

Figure 4-9

SIMULATE−3/CECOR Radial Power Comparison

Cycle 12 at 915 MWD/MTU, 99.5% Power, 1,002 ppm

Axially Integrated



SIMULATE Case Values:
- Average = 1.0073
- RMS Error = 2.5535
- Maximum Value = 1.3320 at Detector 6
- Minimum Value = 0.2980 at Detector 11

CECOR Case Values:
- Average = 1.0072
- Standard Deviation = 0.3019
- Maximum Value = 1.3749 at Detector 23
- Minimum Value = 0.2864 at Detector 11

Absolute Differences:
- Average = 0.0002
- Standard Deviation = 0.0260
- Maximum Value = 0.0595 at Detector 8
- Minimum Value = −0.0429 at Detector 23

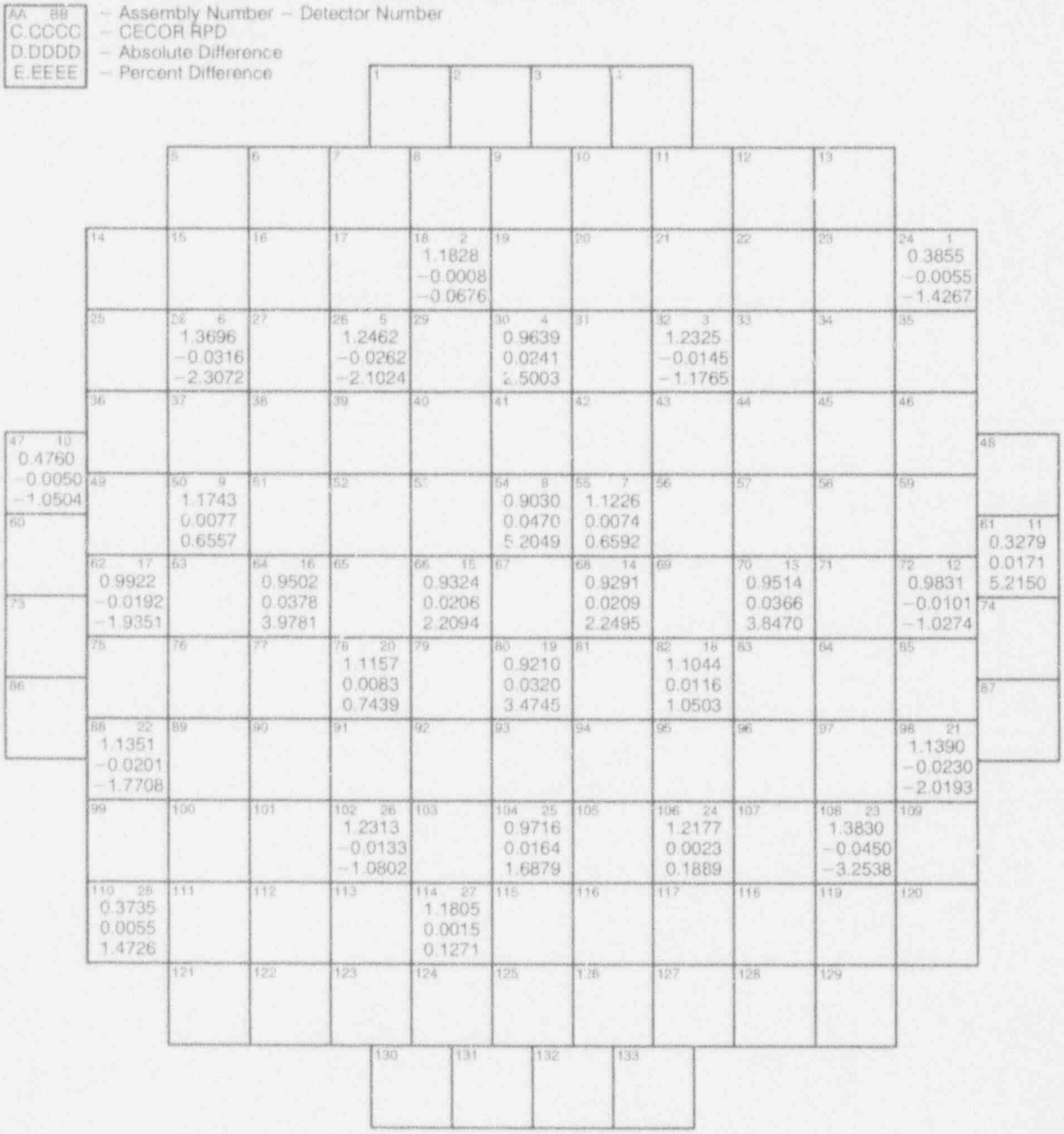
Percentage Differences:
- Average = 0.0518
- Standard Deviation = 2.9007
- Maximum Value = 6.4429 at Detector 8
- Minimum Value = −5.9224 at Detector 1

Figure 4-10

SIMULATE-3/CECOR Radial Power Comparison

Cycle 12 at 5,914 MWD/MTU, 99.7% Power, 614 ppm

Axially Integrated



SIMULATE Case Values:
- Average = 0.9992
- RMS Error = 2.2294
- Maximum Value = 1.3380 at Detector 6
- Minimum Value = 0.3450 at Detector 11

CECOR Case Values:
- Average = 0.9963
- Standard Deviation = 0.2855
- Maximum Value = 1.3830 at Detector 23
- Minimum Value = 0.3279 at Detector 11

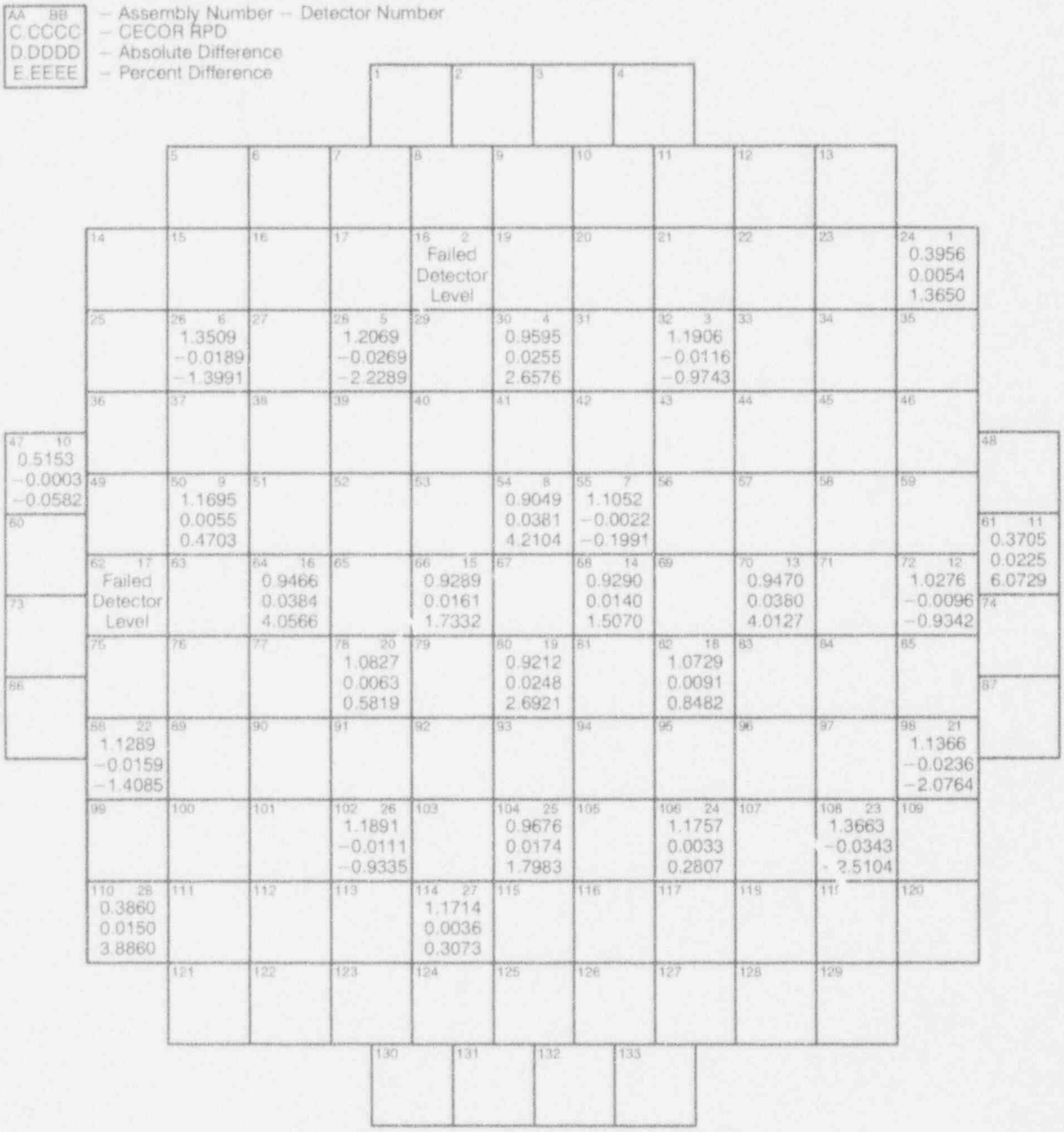
Absolute Differences:
- Average = 0.0029
- Standard Deviation = 0.0225
- Maximum Value = 0.0470 at Detector 8
- Minimum Value = -0.0450 at Detector 23

Percentage Differences:
- Average = 0.5731
- Standard Deviation = 2.3539
- Maximum Value = 5.2150 at Detector 11
- Minimum Value = -3.2538 at Detector 23

Figure 4-11

SIMULATE–3/CECOR Radial Power Comparison

Cycle 12 at 10,931 MWD/MTU, 99.7% Power, 200 ppm
Axially Integrated



SIMULATE Case Values:
- Average = 0.9875
- RMS Error = 2.0336
- Maximum Value = 1.3320 at Detector 6
- Minimum Value = 0.3930 at Detector 11

CECOR Case Values:
- Average = 0.9826
- Standard Deviation = 0.2800
- Maximum Value = 1.3663 at Detector 23
- Minimum Value = 0.3705 at Detector 11

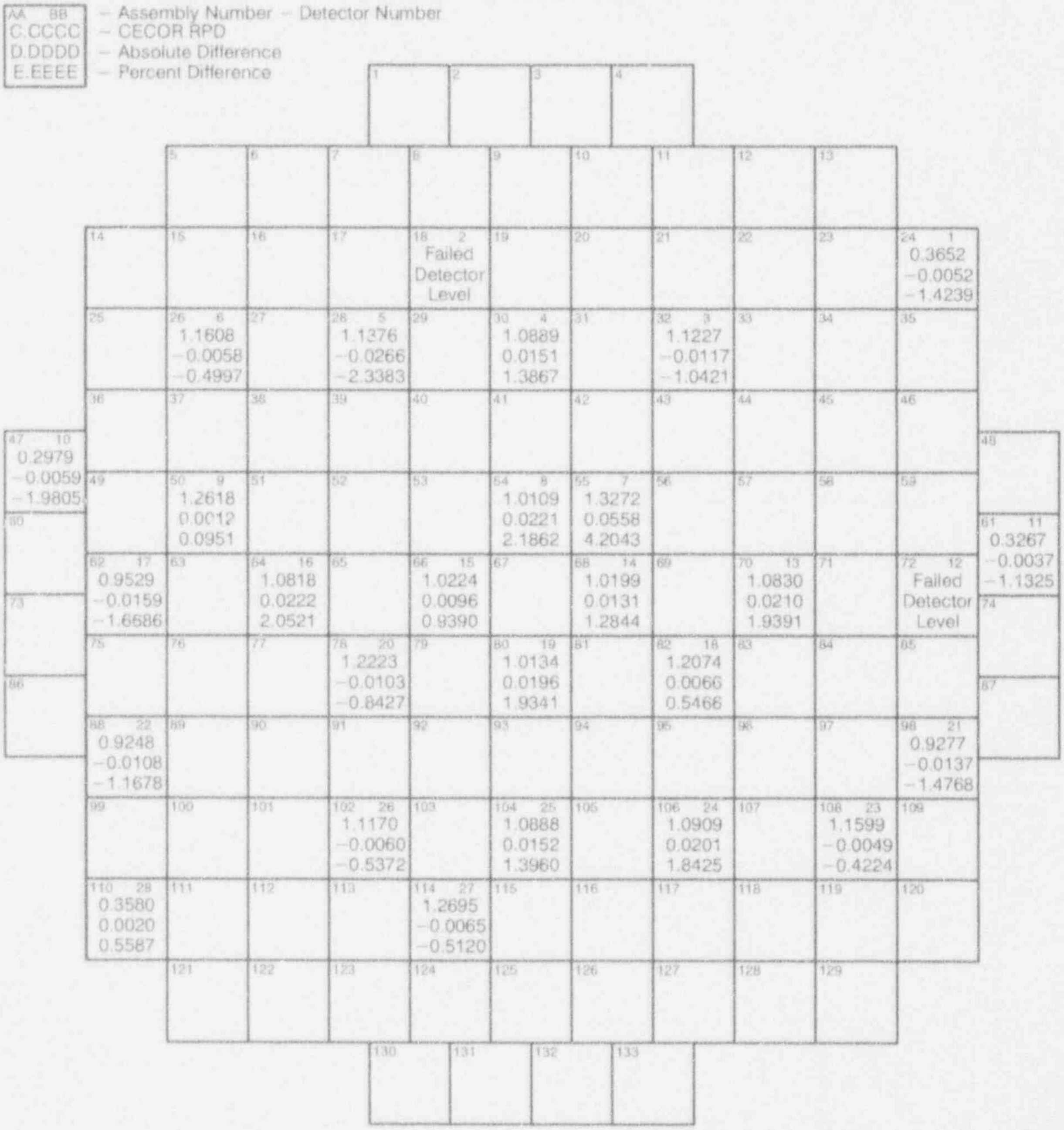
Absolute Differences:
- Average = 0.0049
- Standard Deviation = 0.0201
- Maximum Value = 0.0384 at Detector 16
- Minimum Value = −0.0343 at Detector 23

Percentage Differences:
- Average = 0.9138
- Standard Deviation = 2.2711
- Maximum Value = 6.0729 at Detector 11
- Minimum Value = −2.5104 at Detector 23

Figure 4-12

SIMULATE–3/CECOR Radial Power Comparison

Cycle 13 at 978 MWD/MTU, 99.7% Power, 1,061 ppm
Axially Integrated



SIMULATE Case Values:
- Average = 0.9898
- RMS Error = 1.7337
- Maximum Value = 1.3330 at Detector 7
- Minimum Value = 0.2920 at Detector 10

Absolute Differences:
- Average = 0.0037
- Standard Deviation = 0.0173
- Maximum Value = 0.0558 at Detector 7
- Minimum Value = −0.0266 at Detector 5

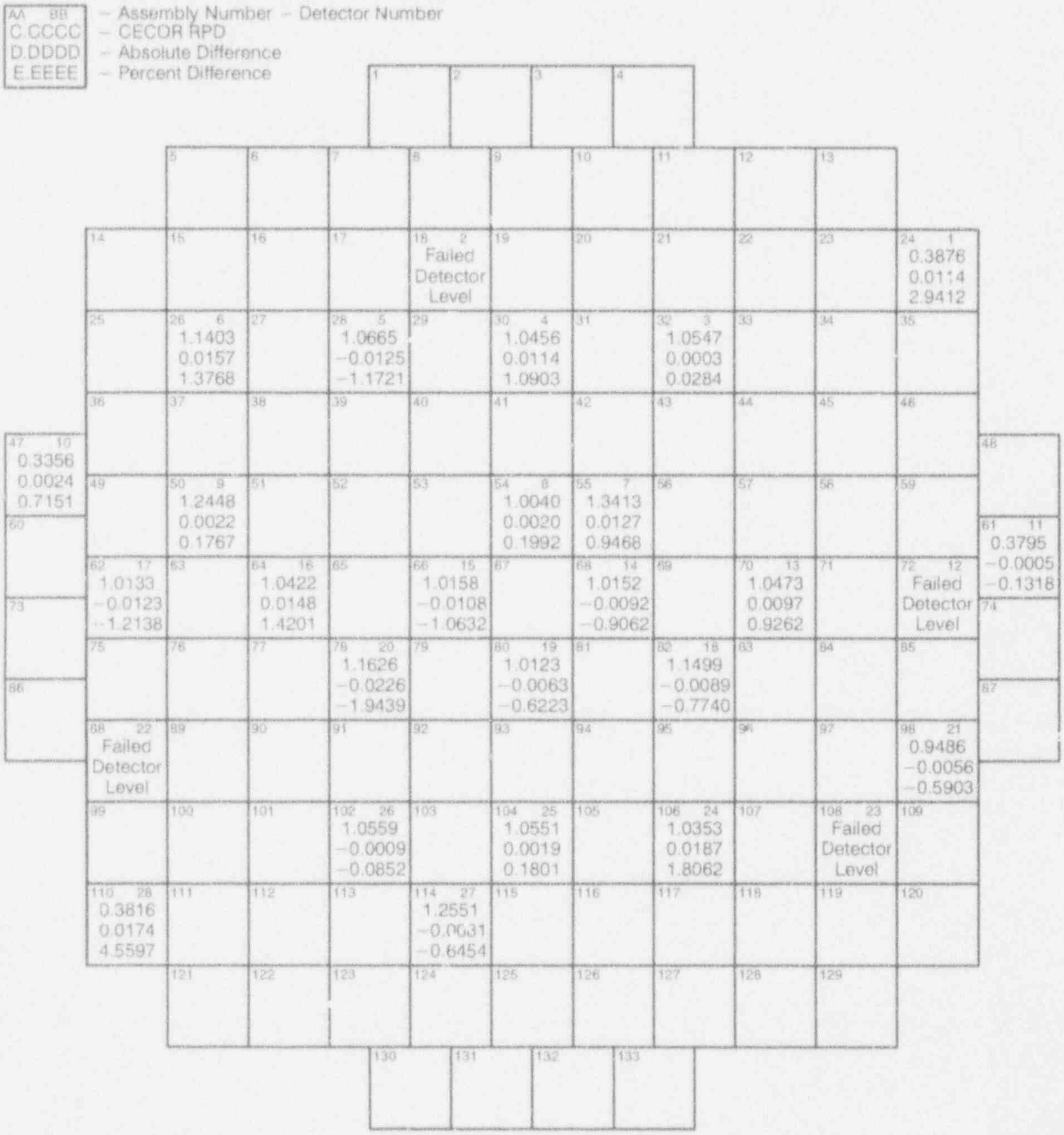
CECOR Case Values:
- Average = 0.9861
- Standard Deviation = 0.3058
- Maximum Value = 1.3272 at Detector 7
- Minimum Value = 0.2979 at Detector 10

Percentage Differences:
- Average = 0.2046
- Standard Deviation = 1.6162
- Maximum Value = 4.2043 at Detector 7
- Minimum Value = −2.3383 at Detector 5

Figure 4-13

SIMULATE – 3/CECOR Radial Power Comparison

Cycle 13 at 7,507 MWD/MTU, 70.2% Power, 629 ppm
Axially Integrated



SIMULATE Case Values:
- Average = 0.9672
- RMS Error = 1.0970
- Maximum Value = 1.3540 at Detector 7
- Minimum Value = 0.3380 at Detector 10

Absolute Differences:
- Average = 0.0010
- Standard Deviation = 0.0112
- Maximum Value = 0.0187 at Detector 24
- Minimum Value = −0.0226 at Detector 20

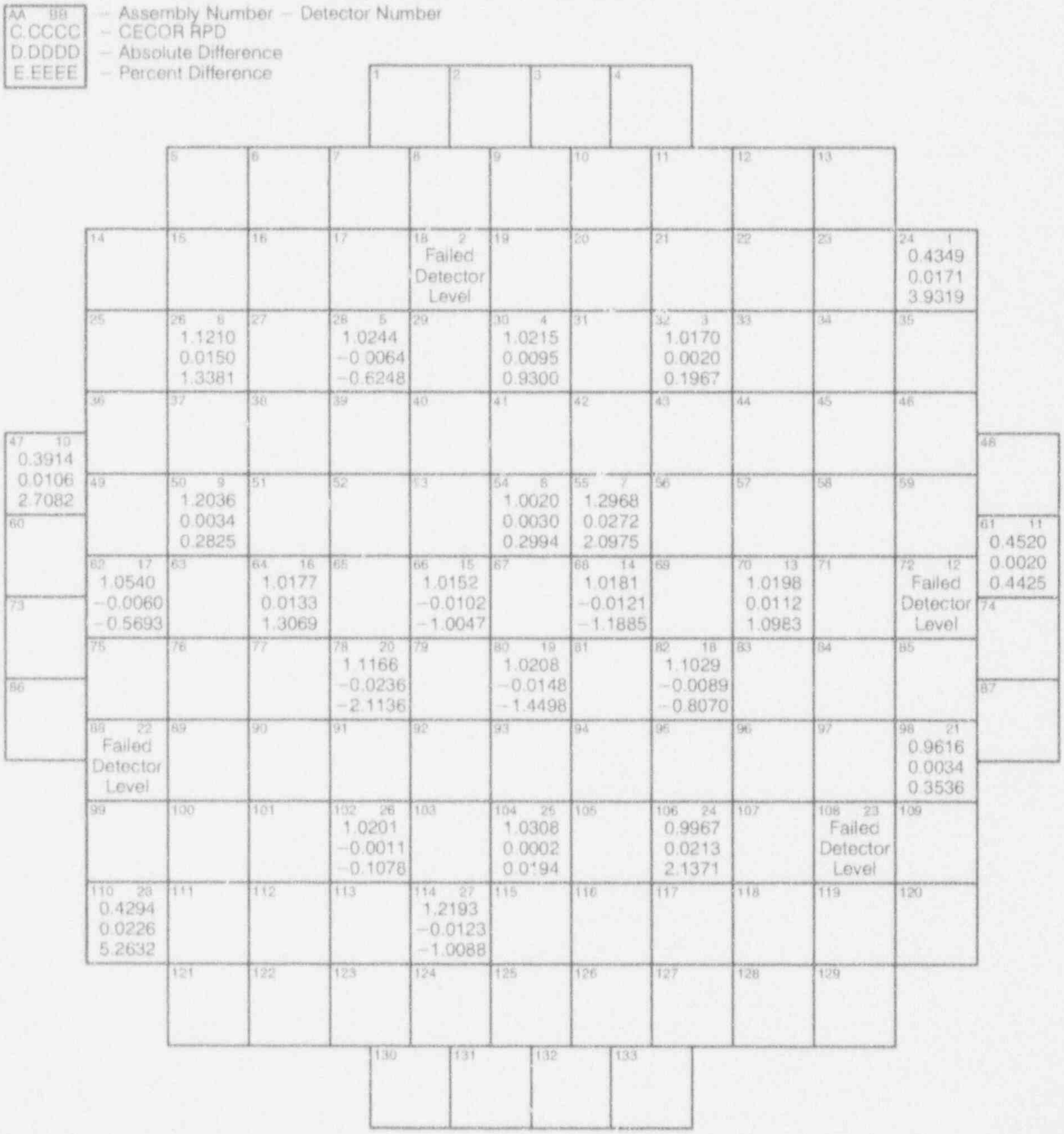
CECOR Case Values:
- Average = 0.9663
- Standard Deviation = 0.2959
- Maximum Value = 1.3413 at Detector 7
- Minimum Value = 0.3356 at Detector 10

Percentage Differences:
- Average = 0.3008
- Standard Deviation = 1.4500
- Maximum Value = 4.5597 at Detector 28
- Minimum Value = −1.9439 at Detector 20

Figure 4-14

SIMULATE–3/CECOR Radial Power Comparison

Cycle 13 at 14,454 MWD/MTU, 99.5% Power, 6 ppm
Axially Integrated



SIMULATE Case Values:
- Average = 0.9606
- RMS Error = 1.3073
- Maximum Value = 1.3240 at Detector 7
- Minimum Value = 0.4020 at Detector 10

CECOR Case Values:
- Average = 0.9578
- Standard Deviation = 0.2641
- Maximum Value = 1.2968 at Detector 7
- Minimum Value = 0.3914 at Detector 10

Absolute Differences:
- Average = 0.0028
- Standard Deviation = 0.0131
- Maximum Value = 0.0272 at Detector 7
- Minimum Value = −0.0236 at Detector 20

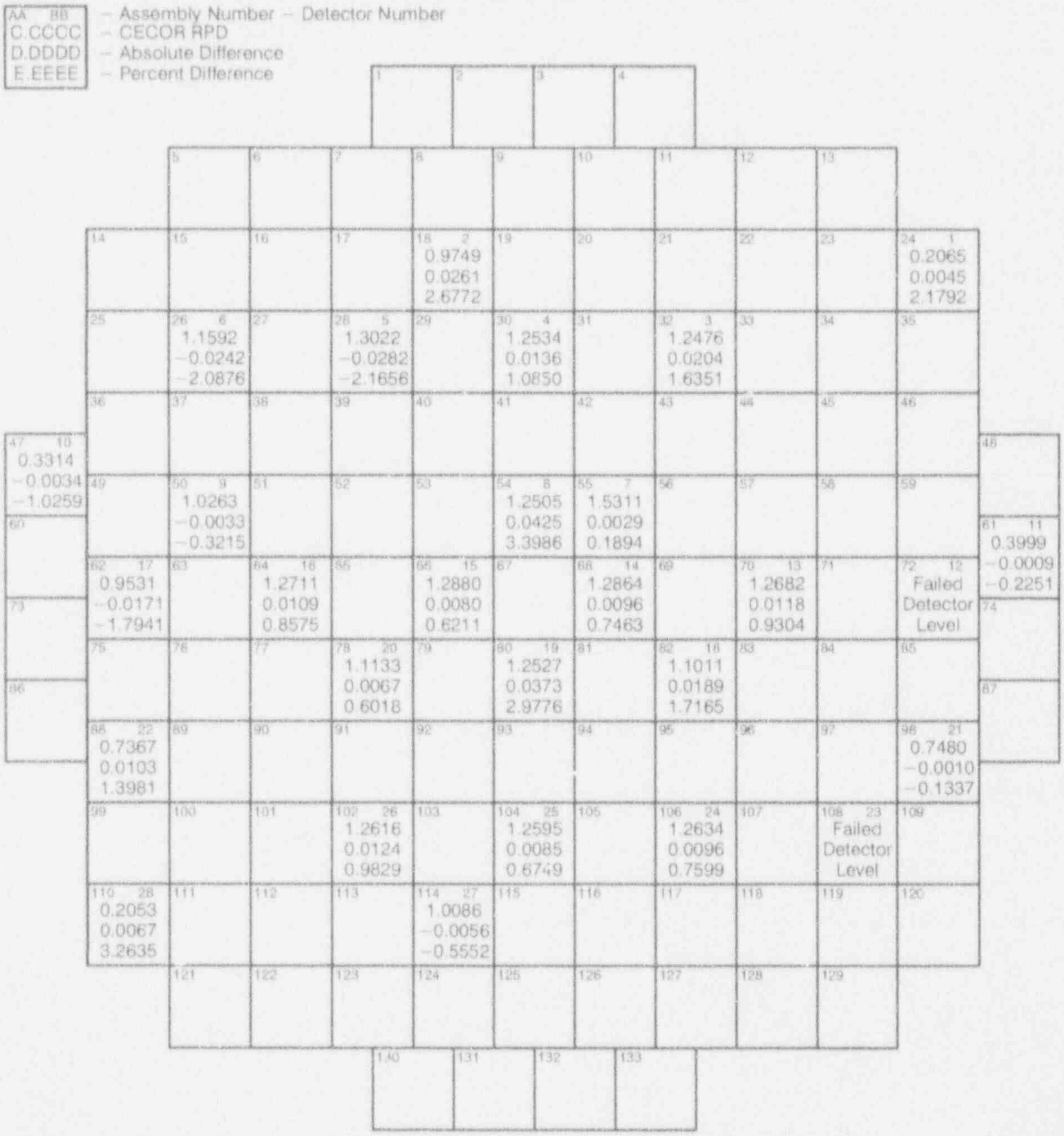
Percentage Differences:
- Average = 0.5638
- Standard Deviation = 1.7360
- Maximum Value = 5.2632 at Detector 28
- Minimum Value = −2.1136 at Detector 20

Figure 4-15

SIMULATE-3 CECOR Radial Power Comparison

Cycle 14 at 1,332 MWD/MTU, 99.6% Power, 780 ppm

Axially Integrated



SIMULATE Case Values:
- Average = 1.0337
- RMS Error = 1.7010
- Maximum Value = 1.5340 at Detector 7
- Minimum Value = 0.2110 at Detector 1

CECOR Case Values:
- Average = 1.0269
- Standard Deviation = 0.3686
- Maximum Value = 1.5311 at Detector 7
- Minimum Value = 0.2053 at Detector 28

Absolute Differences:
- Average = 0.0068
- Standard Deviation = 0.0159
- Maximum Value = 0.0425 at Detector 8
- Minimum Value = -0.0282 at Detector 5

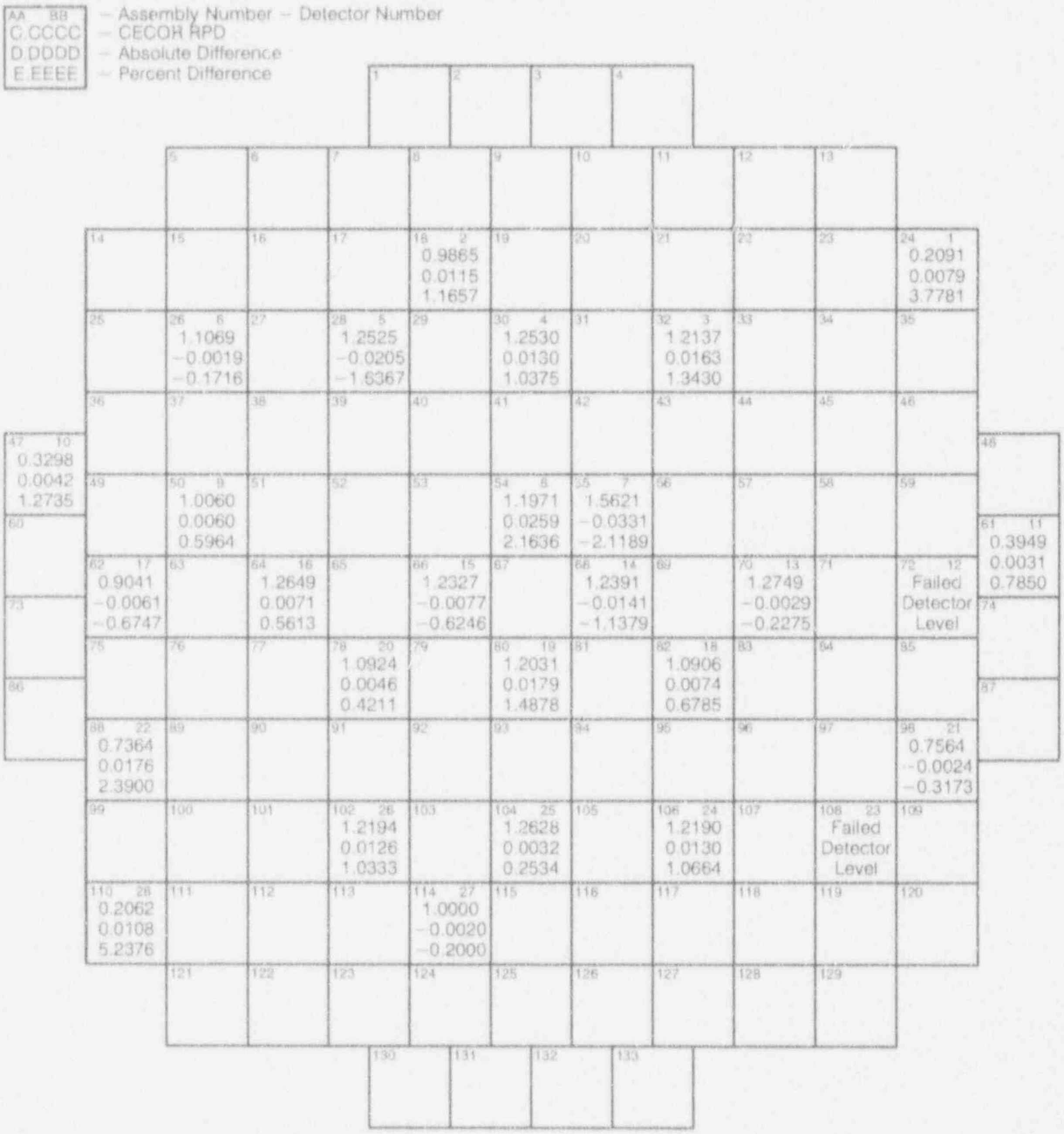
Percentage Differences:
- Average = 0.7072
- Standard Deviation = 1.5162
- Maximum Value = 3.3986 at Detector 8
- Minimum Value = -2.1656 at Detector 5

Figure 4-16

SIMULATE–3/CECOR Radial Power Comparison

Cycle 14 at 7,125 MWD/MTU, 99.8% Power, 603 ppm

Axially Integrated



SIMULATE Case Values:
- Average = 1.0117
- RMS Error = 1.3020
- Maximum Value = 1.5290 at Detector 7
- Minimum Value = 0.2170 at Detector 28

CECOR Case Values:
- Average = 1.0082
- Standard Deviation = 0.3605
- Maximum Value = 1.5621 at Detector 7
- Minimum Value = 0.2062 at Detector 28

Absolute Differences:
- Average = 0.0035
- Standard Deviation = 0.0128
- Maximum Value = 0.0259 at Detector 8
- Minimum Value = −0.0331 at Detector 7

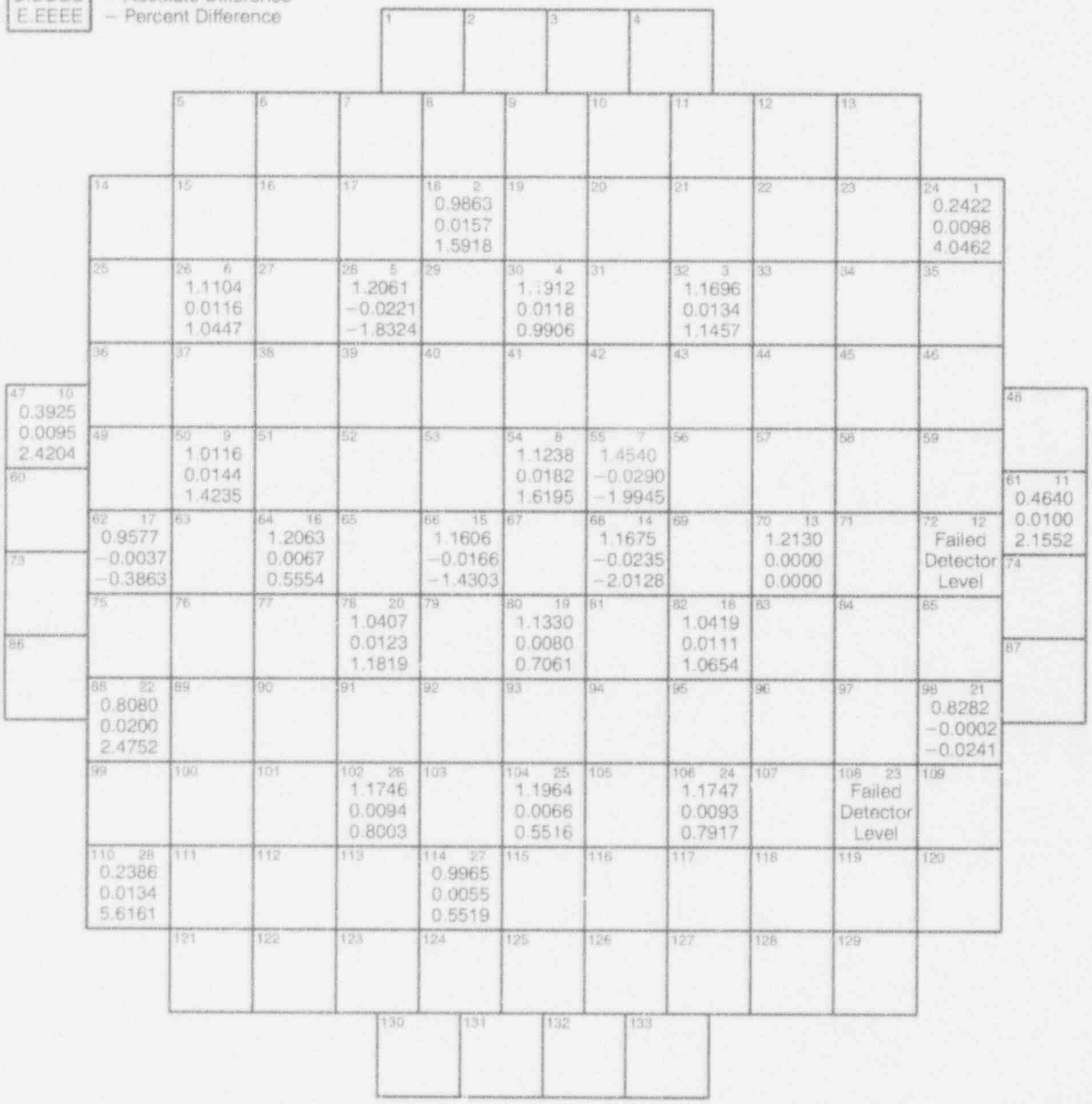
Percentage Differences:
- Average = 0.6986
- Standard Deviation = 1.5647
- Maximum Value = 5.2376 at Detector 28
- Minimum Value = −2.1189 at Detector 7

Figure 4-17

SIMULATE–3/CECOR Radial Power Comparison

Cycle 14 at 13,316 MWD/MTU, 99.7% Power, 58 ppm
Axially Integrated

AA BB	– Assembly Number – Detector Number
C.CCCC	– CECOR RPD
D.DDDD	– Absolute Difference
E.EEEE	– Percent Difference



SIMULATE Case Values:
- Average = 0.9927
- RMS Error = 1.3738
- Maximum Value = 1.4250 at Detector 7
- Minimum Value = 0.2520 at Detector 28

CECOR Case Values:
- Average = 0.9881
- Standard Deviation = 0.3159
- Maximum Value = 1.4540 at Detector 7
- Minimum Value = 0.2386 at Detector 28

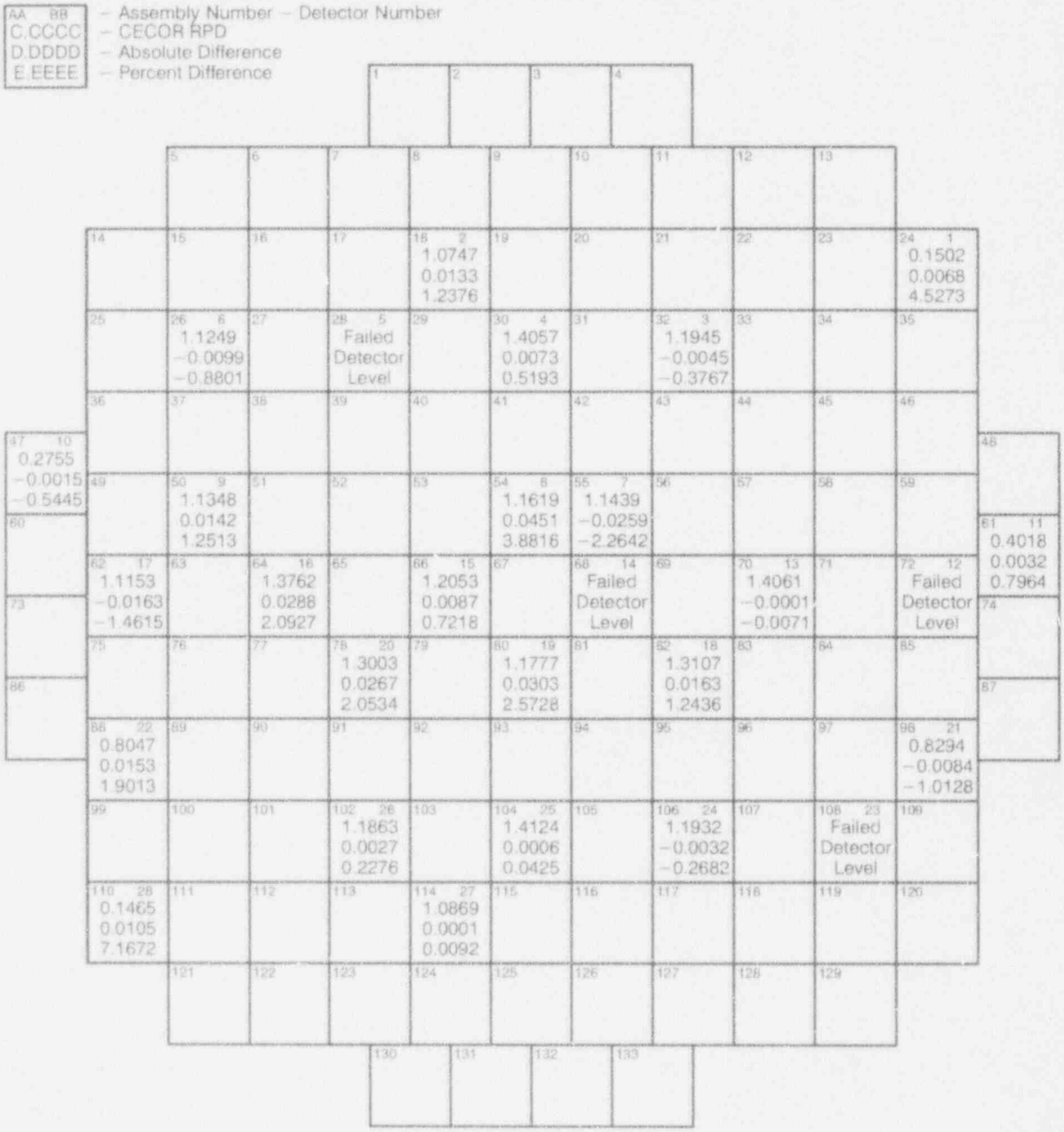
Absolute Differences:
- Average = 0.0047
- Standard Deviation = 0.0132
- Maximum Value = 0.0200 at Detector 22
- Minimum Value = −0.0290 at Detector 7

Percentage Differences:
- Average = 0.8866
- Standard Deviation = 1.7143
- Maximum Value = 5.6161 at Detector 28
- Minimum Value = −2.0128 at Detector 14

Figure 4-18

SIMULATE–3/CECOR Radial Power Comparison

Cycle 15 at 1,114 MWD/MTU, 99.9% Power, 945 ppm
Axially Integrated



SIMULATE Case Values:
- Average = 1.0325
- RMS Error = 1.6850
- Maximum Value = 1.4130 at Detector 4
- Minimum Value = 0.1570 at Detector 28

CECOR Case Values:
- Average = 1.0258
- Standard Deviation = 0.3945
- Maximum Value = 1.4124 at Detector 25
- Minimum Value = 0.1465 at Detector 28

Absolute Differences:
- Average = 0.0067
- Standard Deviation = 0.0158
- Maximum Value = 0.0451 at Detector 8
- Minimum Value = −0.0259 at Detector 7

Percentage Differences:
- Average = 0.9763
- Standard Deviation = 2.0703
- Maximum Value = 7.1672 at Detector 28
- Minimum Value = −2.2642 at Detector 7

Figure 4-19

Cycle 11 Axial Power Distribution Comparison

1,094 MWD/MTU, 99.4% Power, 973 ppm

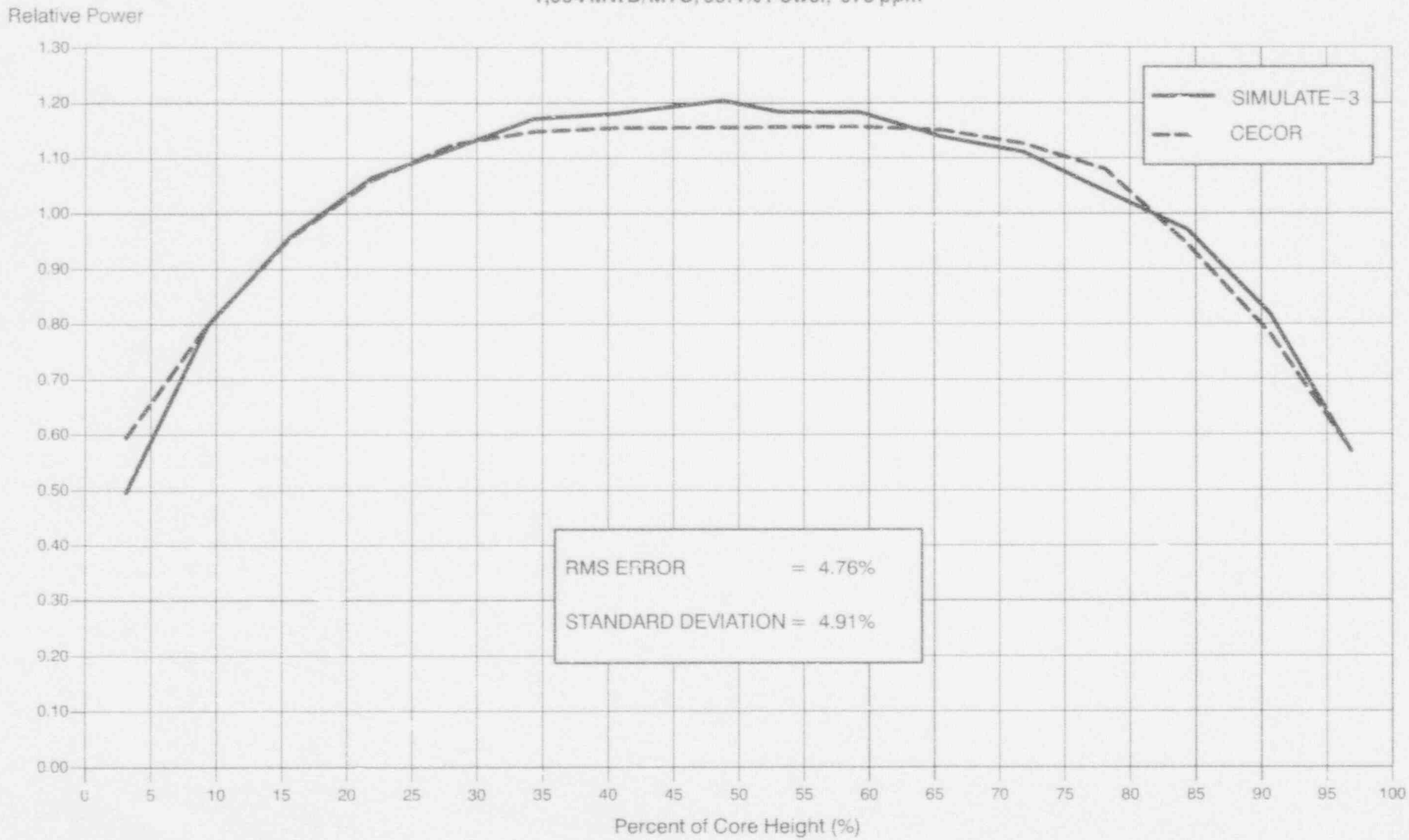


Figure 4-20

Cycle 11 Axial Power Distribution Comparison

6,990 MWD/MTU, 99.5% Power, 532 ppm

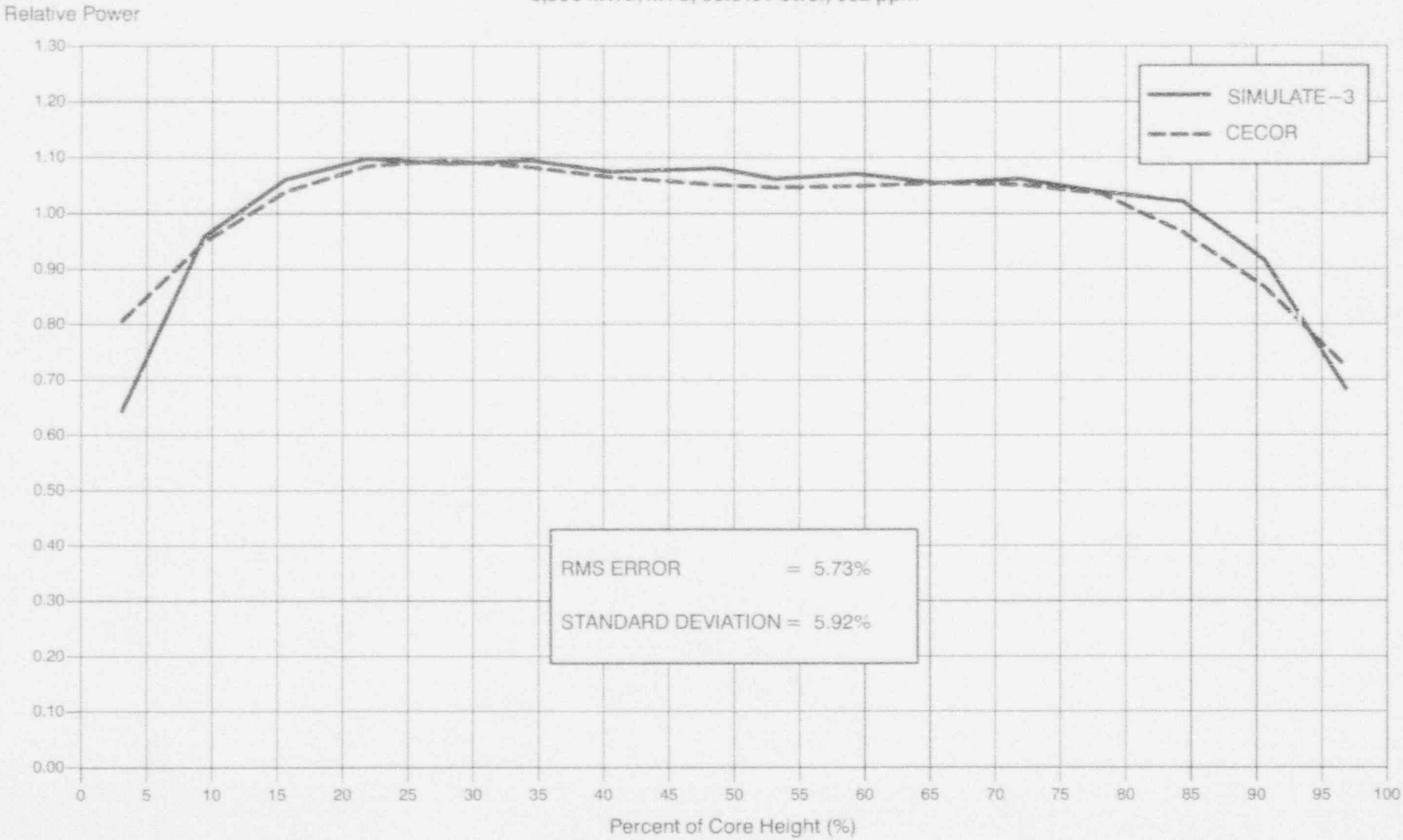


Figure 4-21

Cycle 11 Axial Power Distribution Comparison

11,088 MWD/MTU, 99.3% Power, 189 ppm

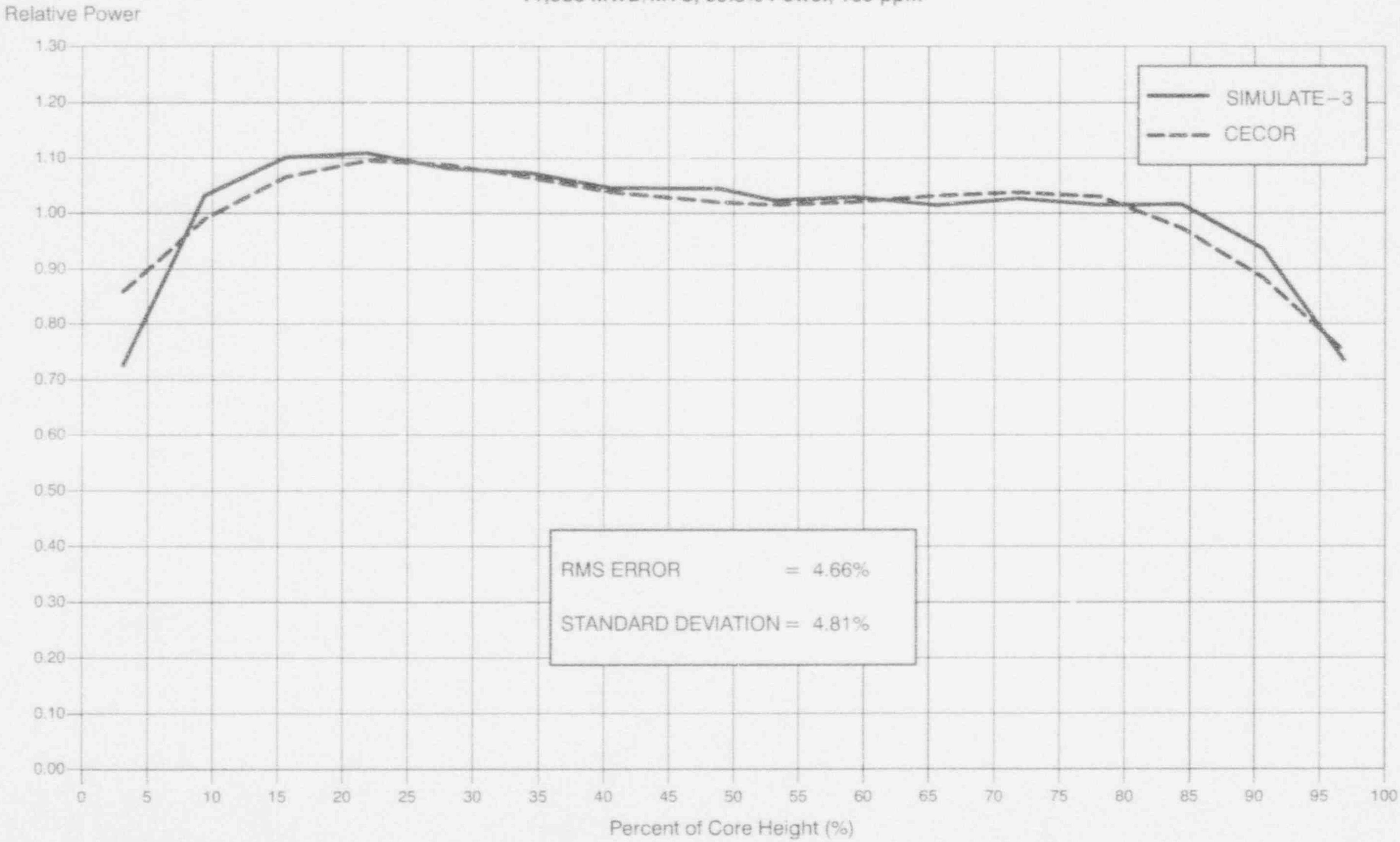


Figure 4-22

Cycle 12 Axial Power Distribution Comparison

915 MWD/MTU, 99.5% Power, 1002. ppm

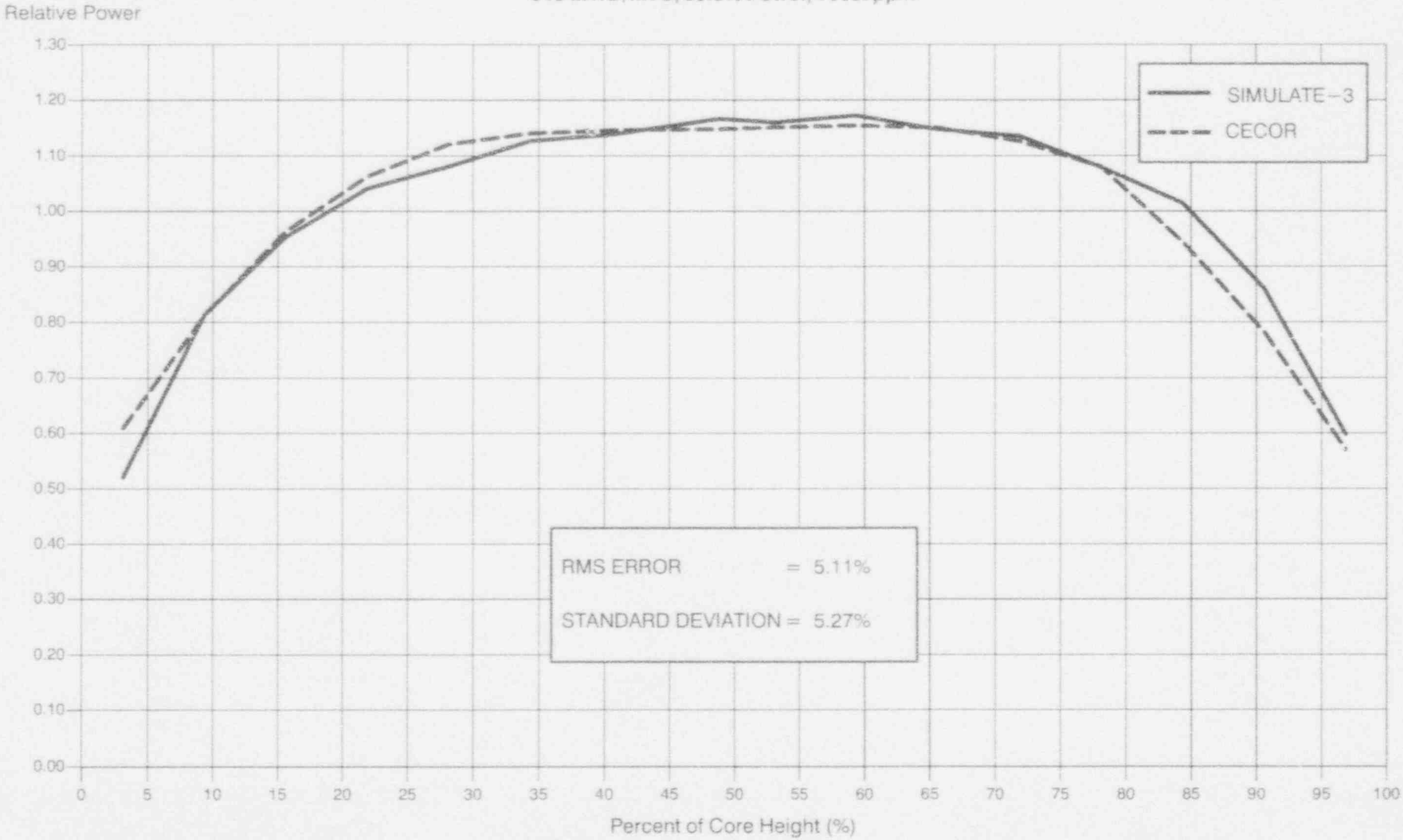


Figure 4-23

Cycle 12 Axial Power Distribution Comparison

5,914 MWD/MTU, 99.7% Power, 614 ppm

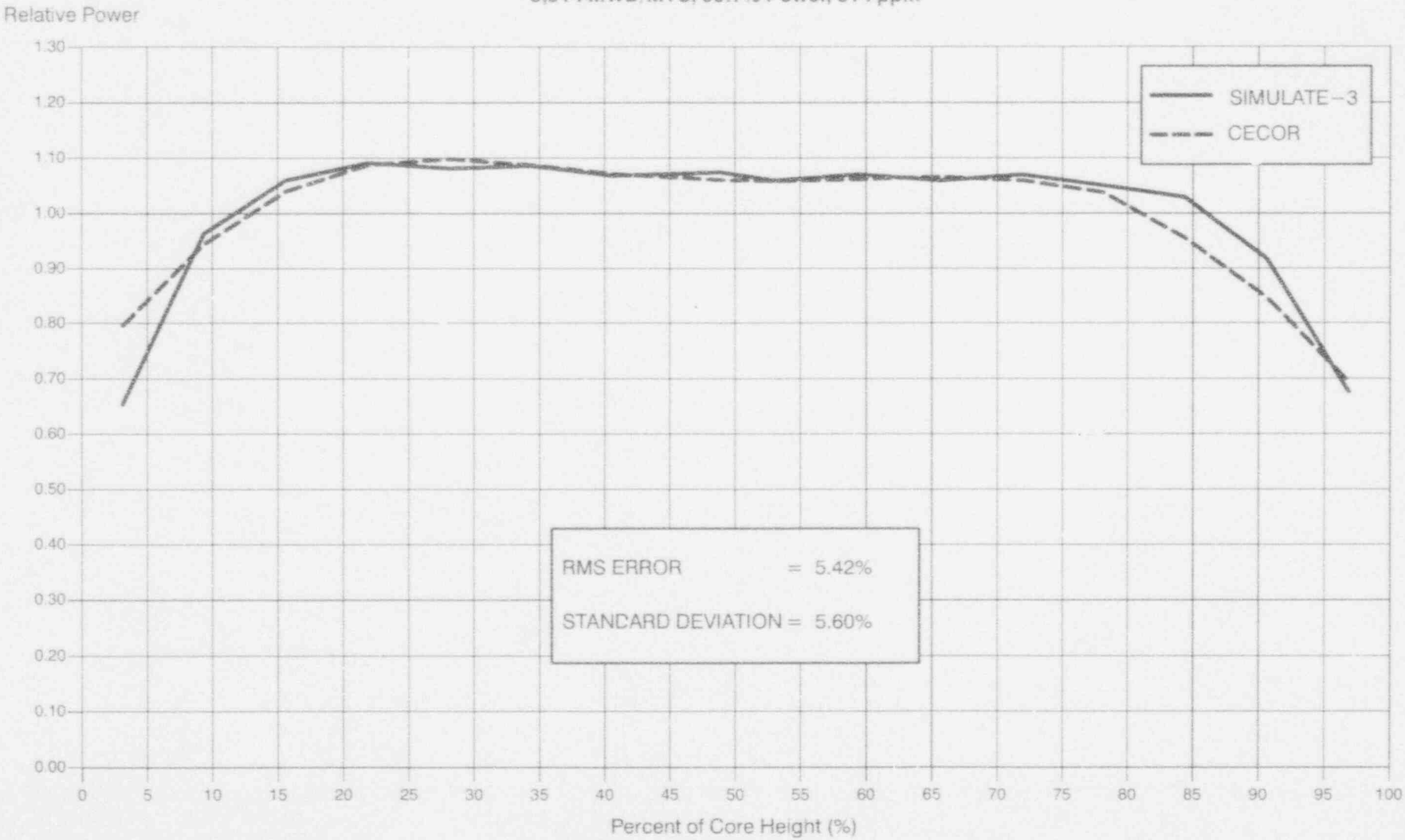


Figure 4-24

Cycle 12 Axial Power Distribution Comparison

10,931 MWD/MTU, 99.7% Power, 200 ppm

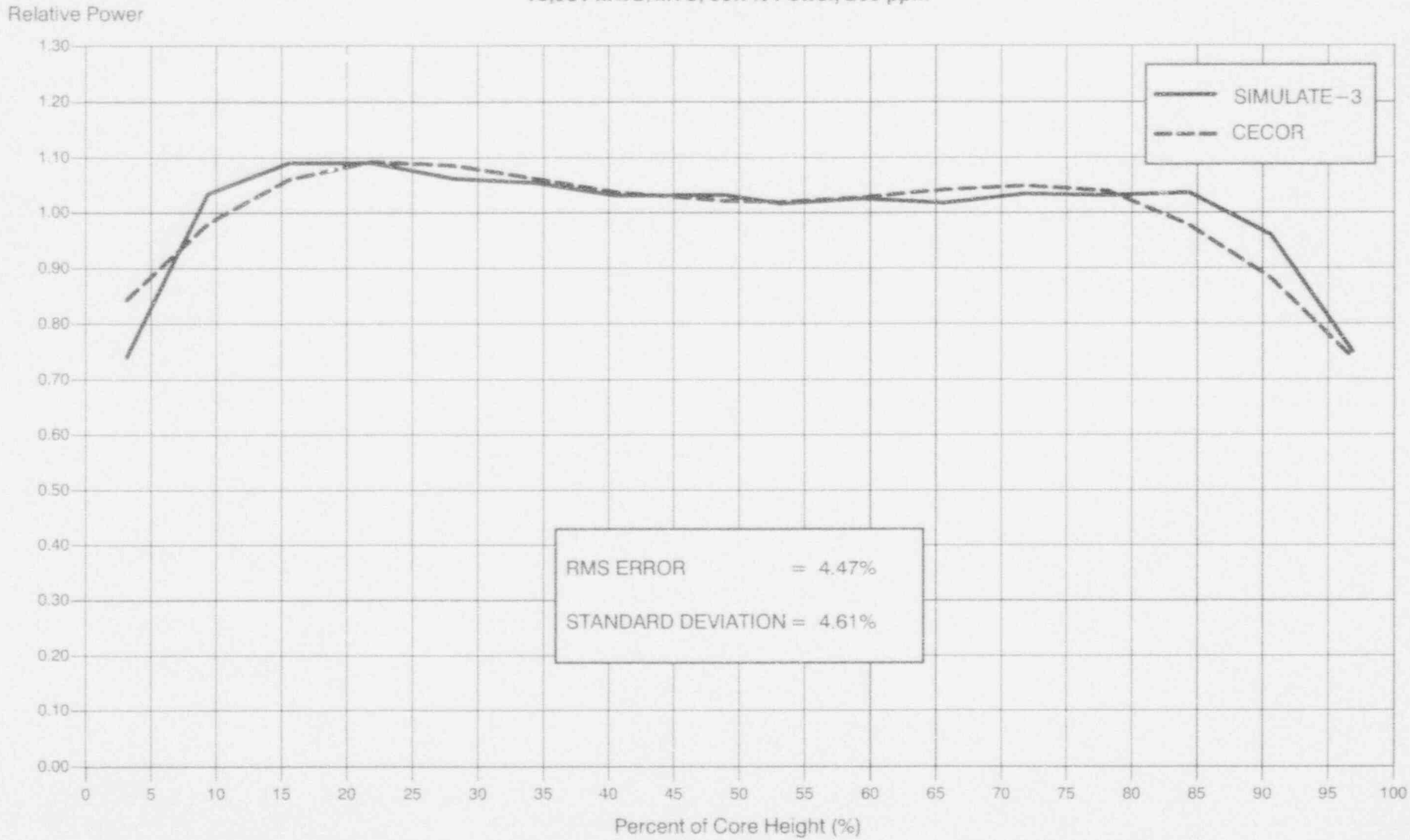


Figure 4-25

Cycle 13 Axial Power Distribution Comparison

978 MWD/MTU, 99.7% Power, 1061 ppm

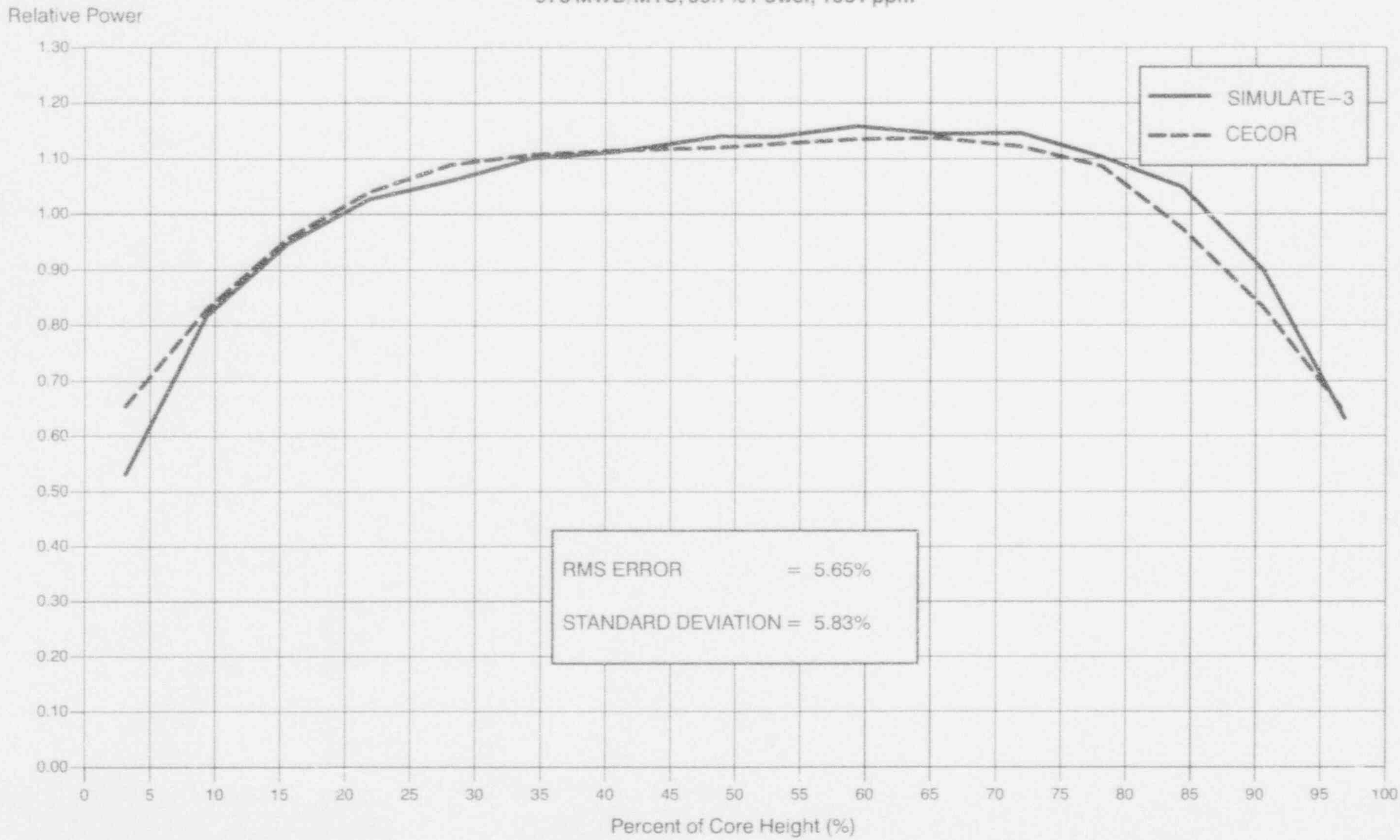


Figure 4-26

Cycle 13 Axial Power Distribution Comparison

7,507 MWD/MTU, 70.2% Power, 629 ppm

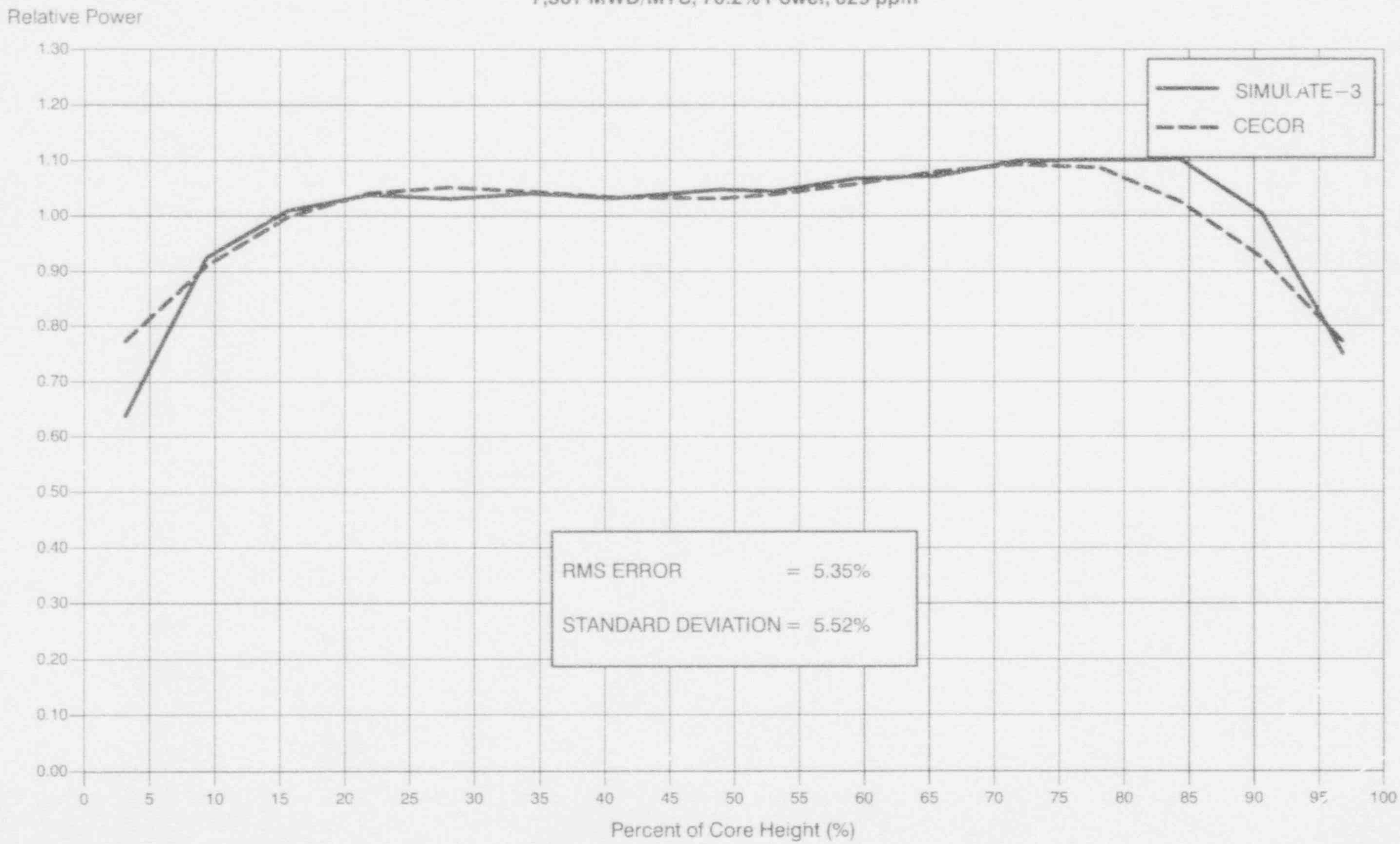


Figure 4-27

Cycle 13 Axial Power Distribution Comparison

14,454 MWD/MTU, 99.5% Power, 6 ppm

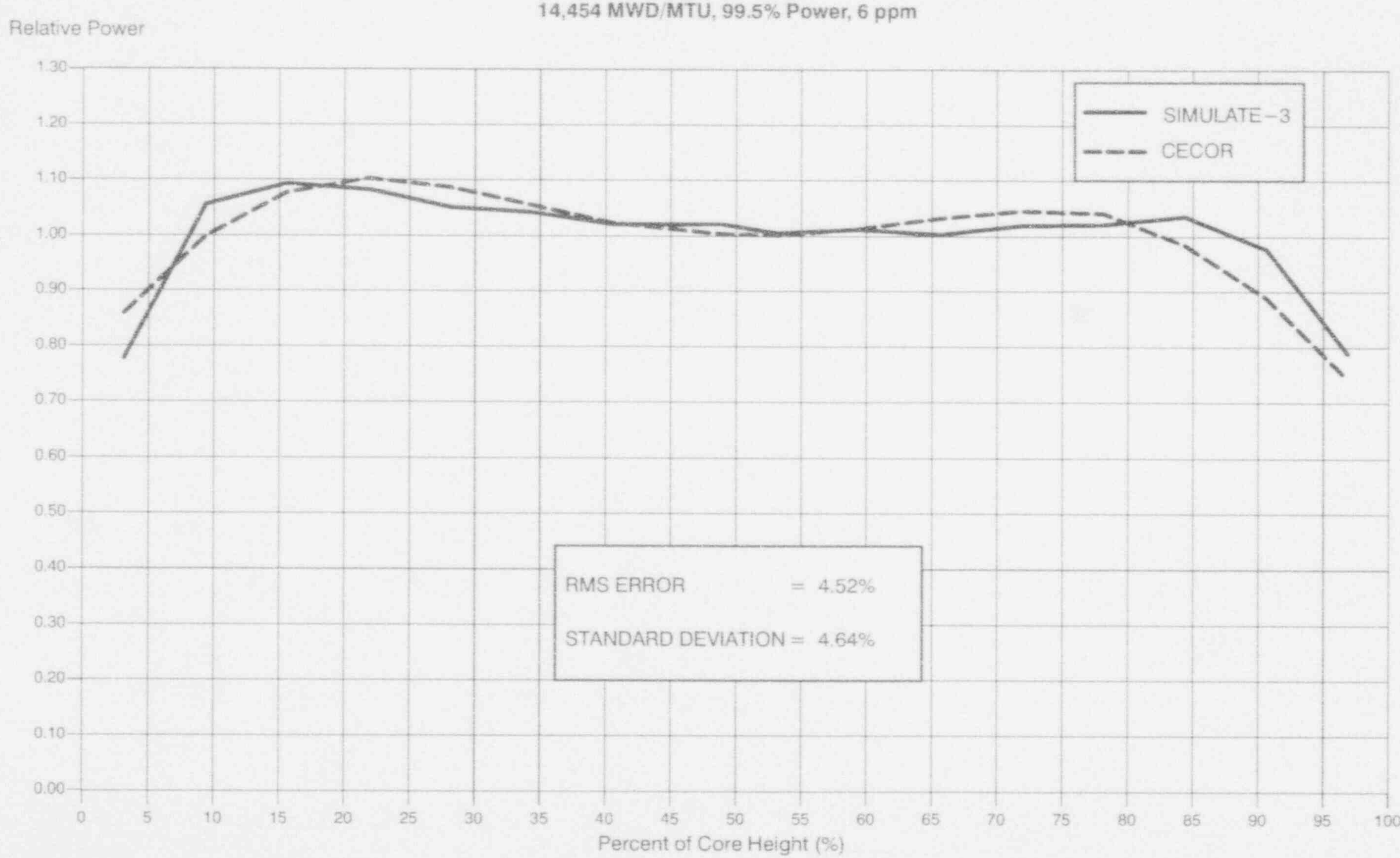


Figure 4-28

Cycle 14 Axial Power Distribution Comparison

1,332 MWD/MTU, 99.6% Power, 780 ppm

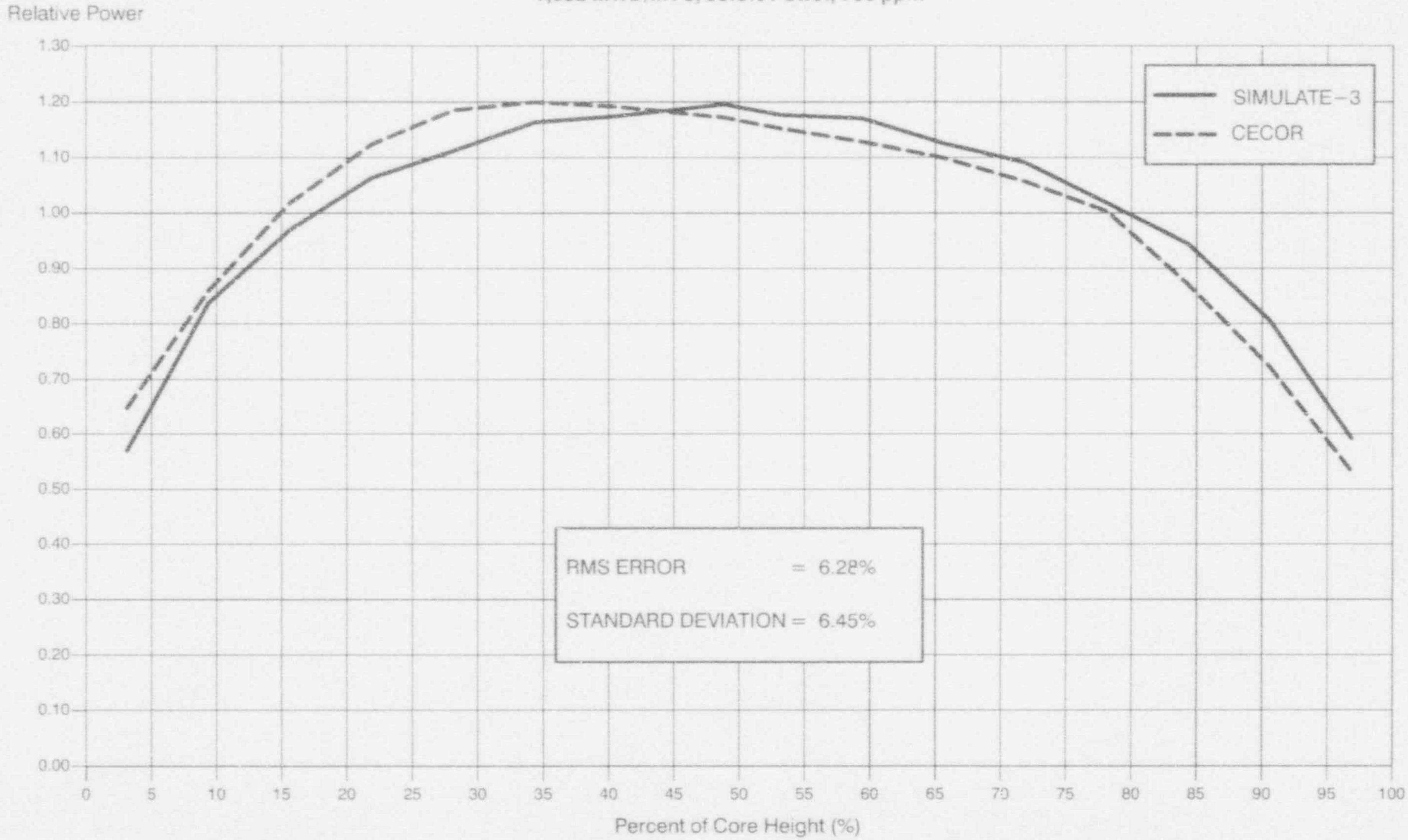


Figure 4-29

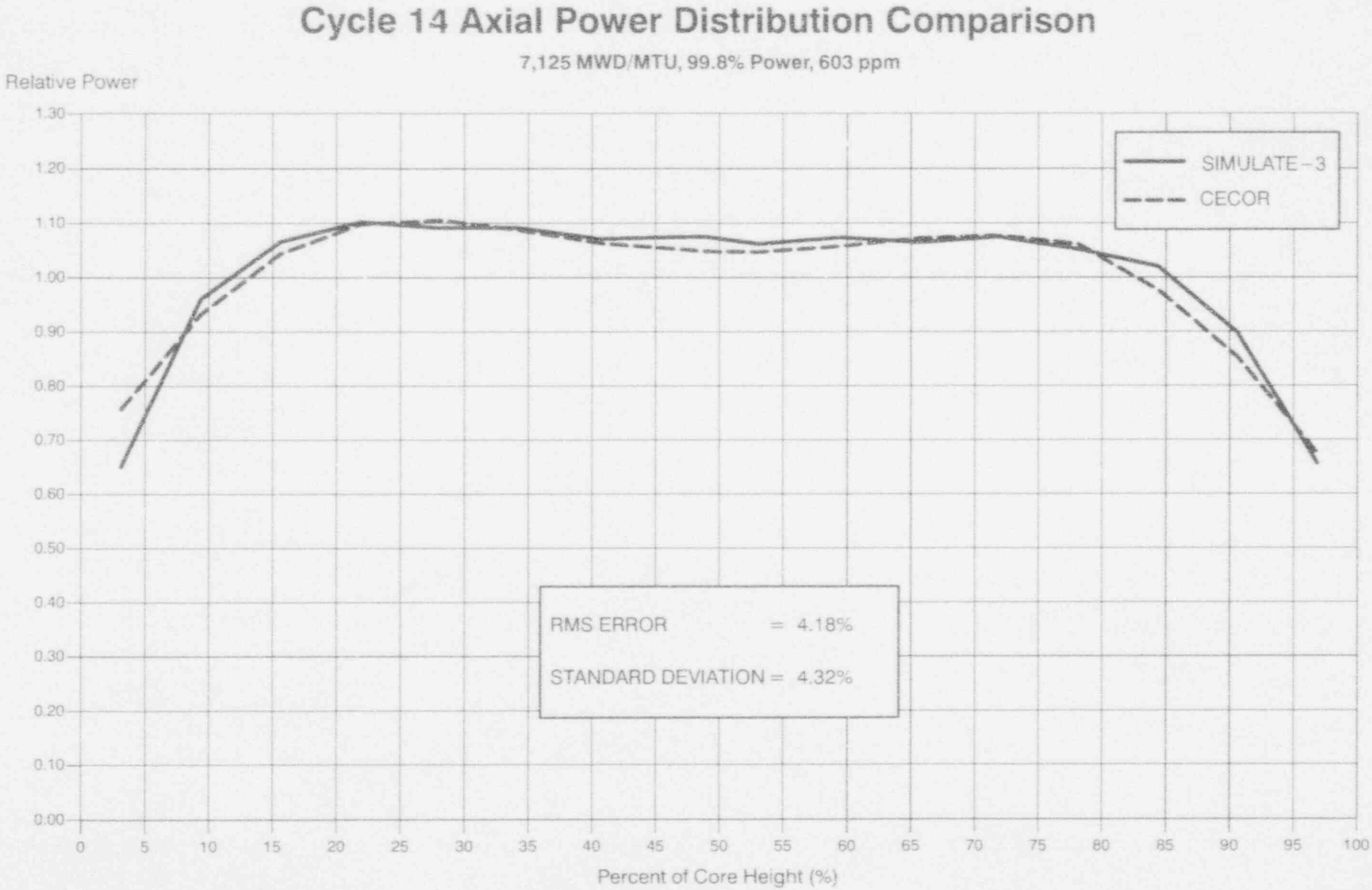

Cycle 14 Axial Power Distribution Comparison
7,125 MWD/MTU, 99.8% Power, 603 ppm
Relative Power
1.30
1.20
1.10
1.00
0.90
0.80
0.70
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
5
10
15
20
25
30
35
40
45
50
55
60
65
70
75
80
85
90
95
100
Percent of Core Height (%)
SIMULATE-3
CECOR
RMS ERROR = 4.18%
STANDARD DEVIATION = 4.32%

Figure 4-30

Cycle 14 Axial Power Distribution Comparison

13,316 MWD/MTU, 99.7% Power, 58 ppm

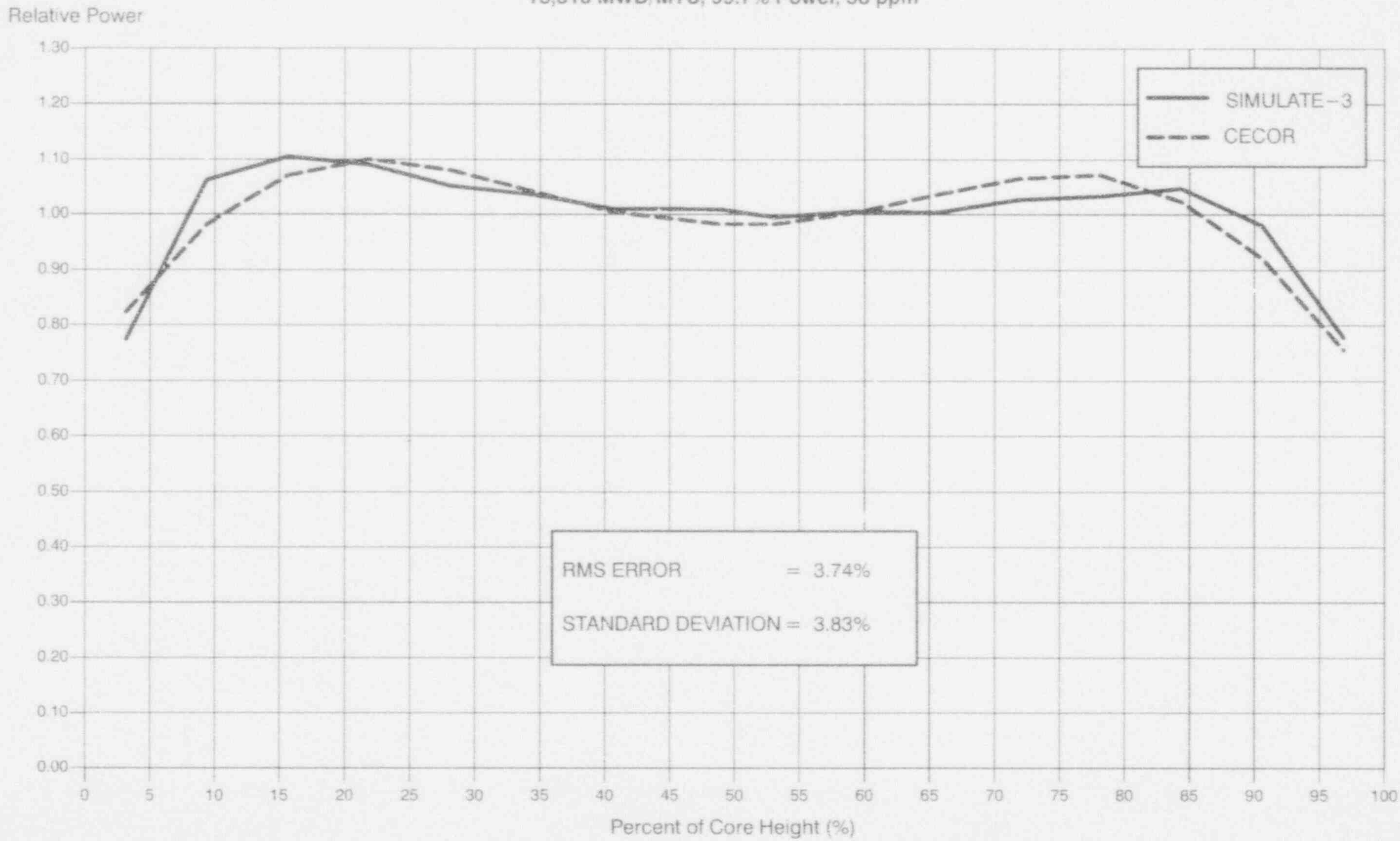


Figure 4-31

Cycle 15 Axial Power Distribution Comparison

1,114 MWD/MTU, 99.9% Power, 945 ppm

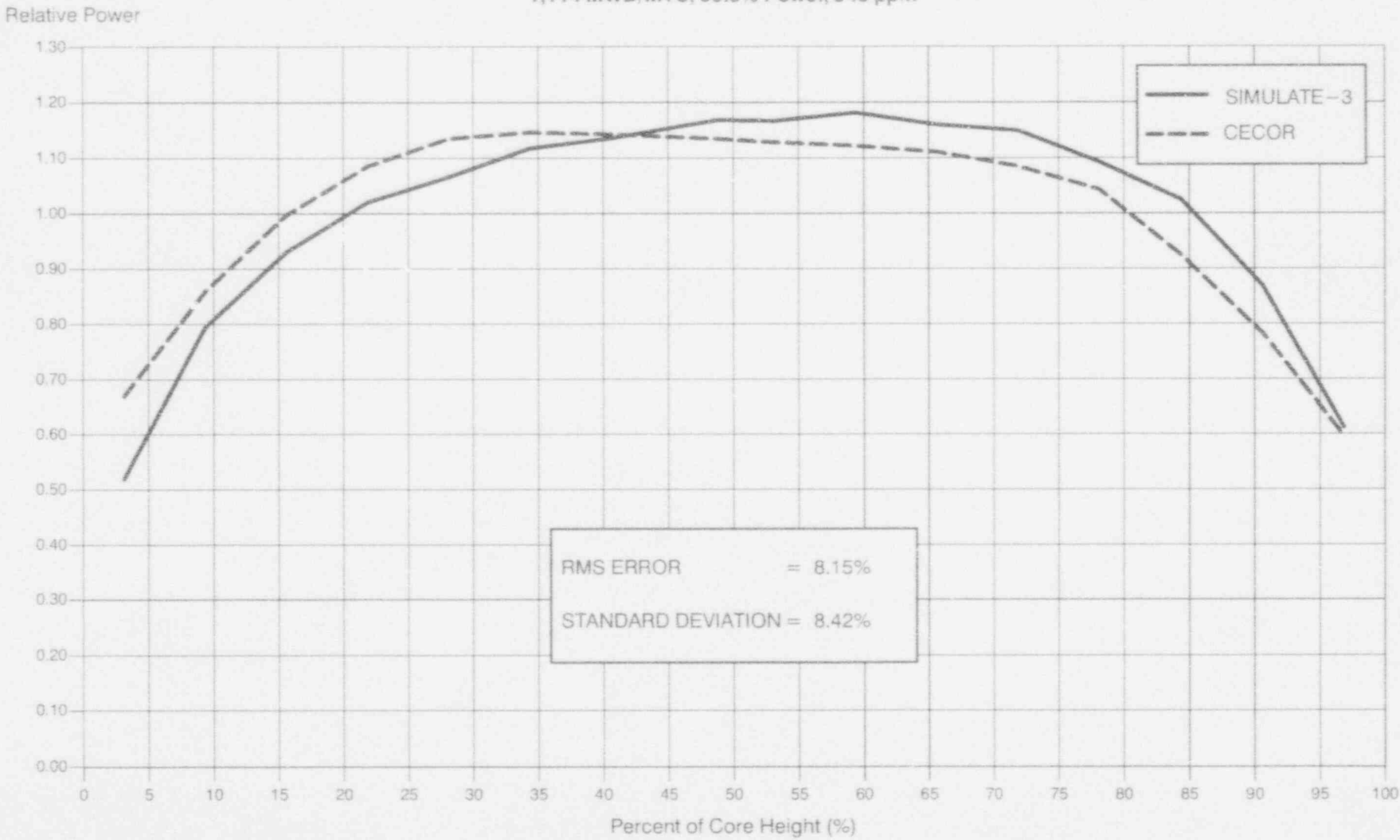


Figure 4-32

Cycle 11 Integrated Radial Peaking (F_R) vs. Burnup

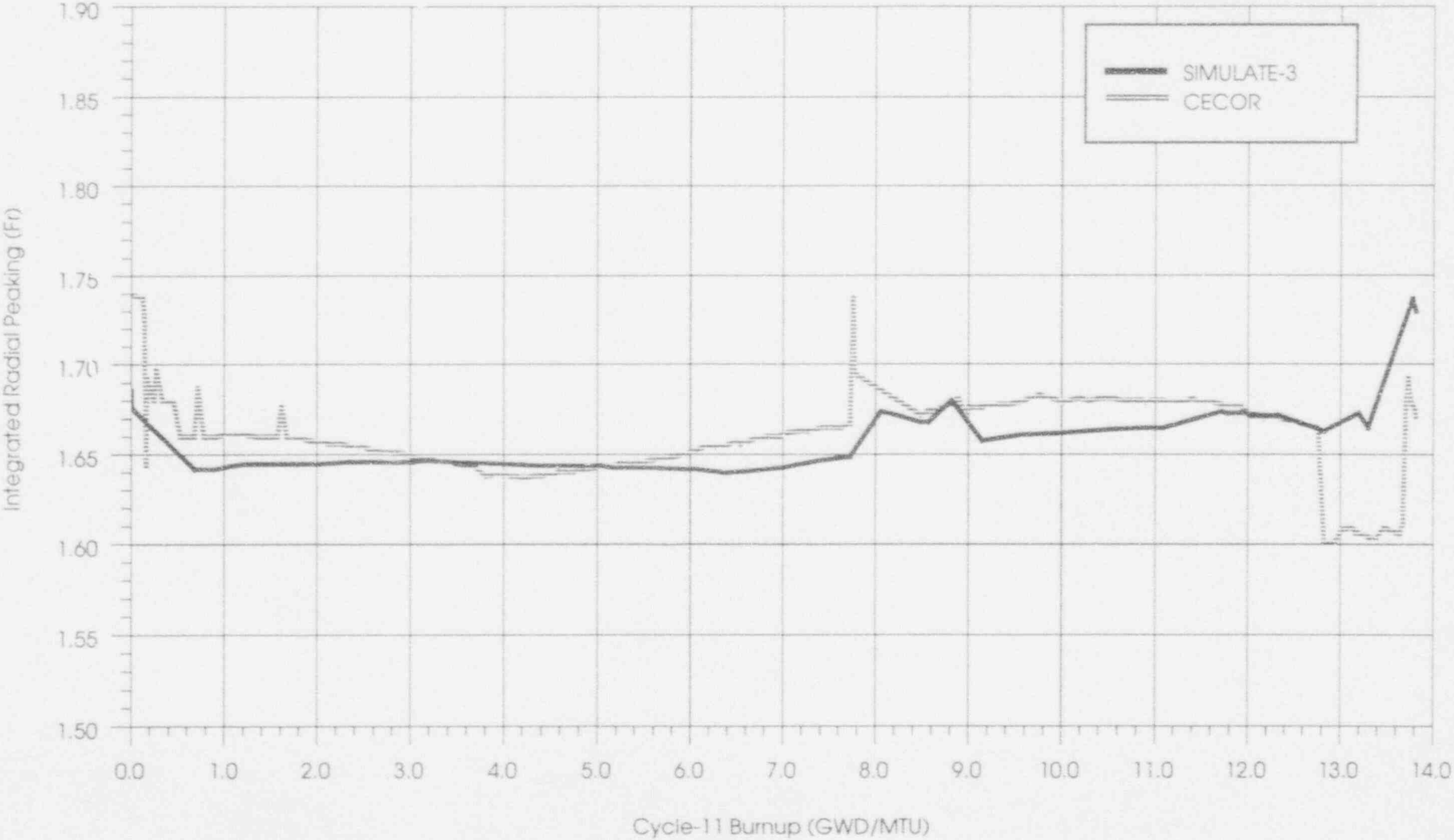


Figure 4-33

Cycle 12 Integrated Radial Peaking (F_R) vs. Burnup

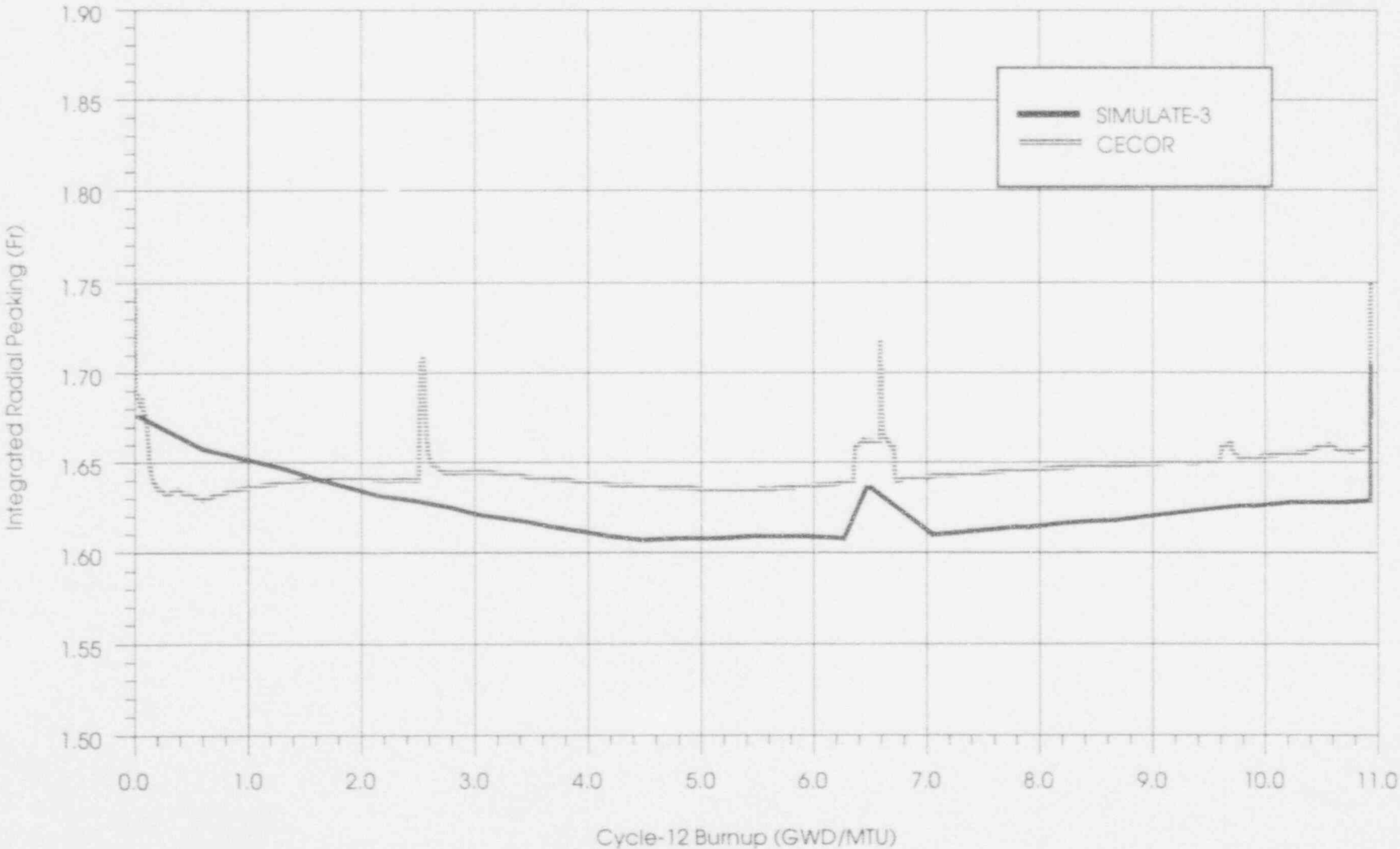


Figure 4-34

Cycle 13 Integrated Radial Peaking (F_R) vs. Burnup

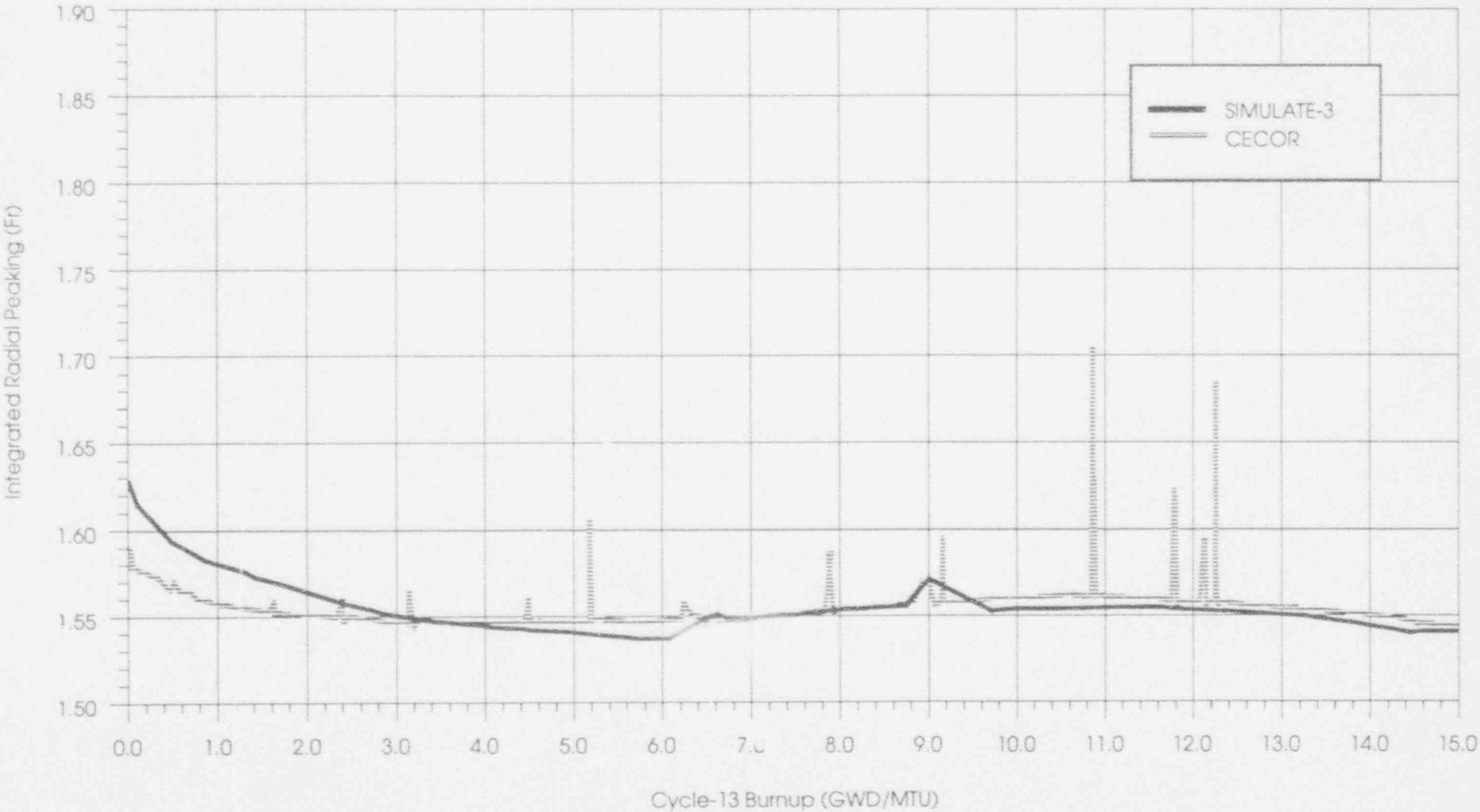


Figure 4-35

Cycle 14 Integrated Radial Peaking (F_R) vs. Burnup

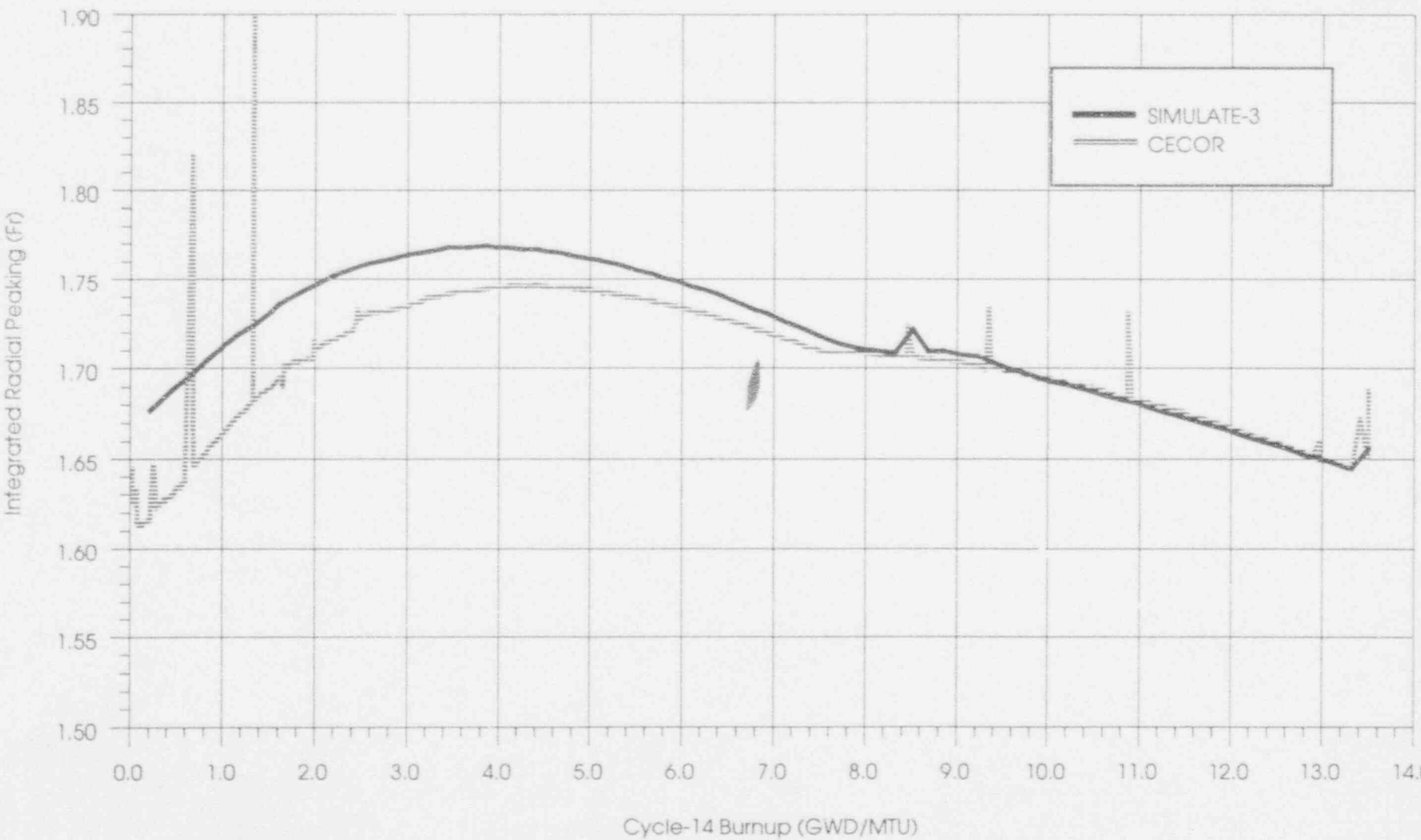


Figure 4-36

Cycle 15 Integrated Radial Peaking (F_R) vs. Burnup

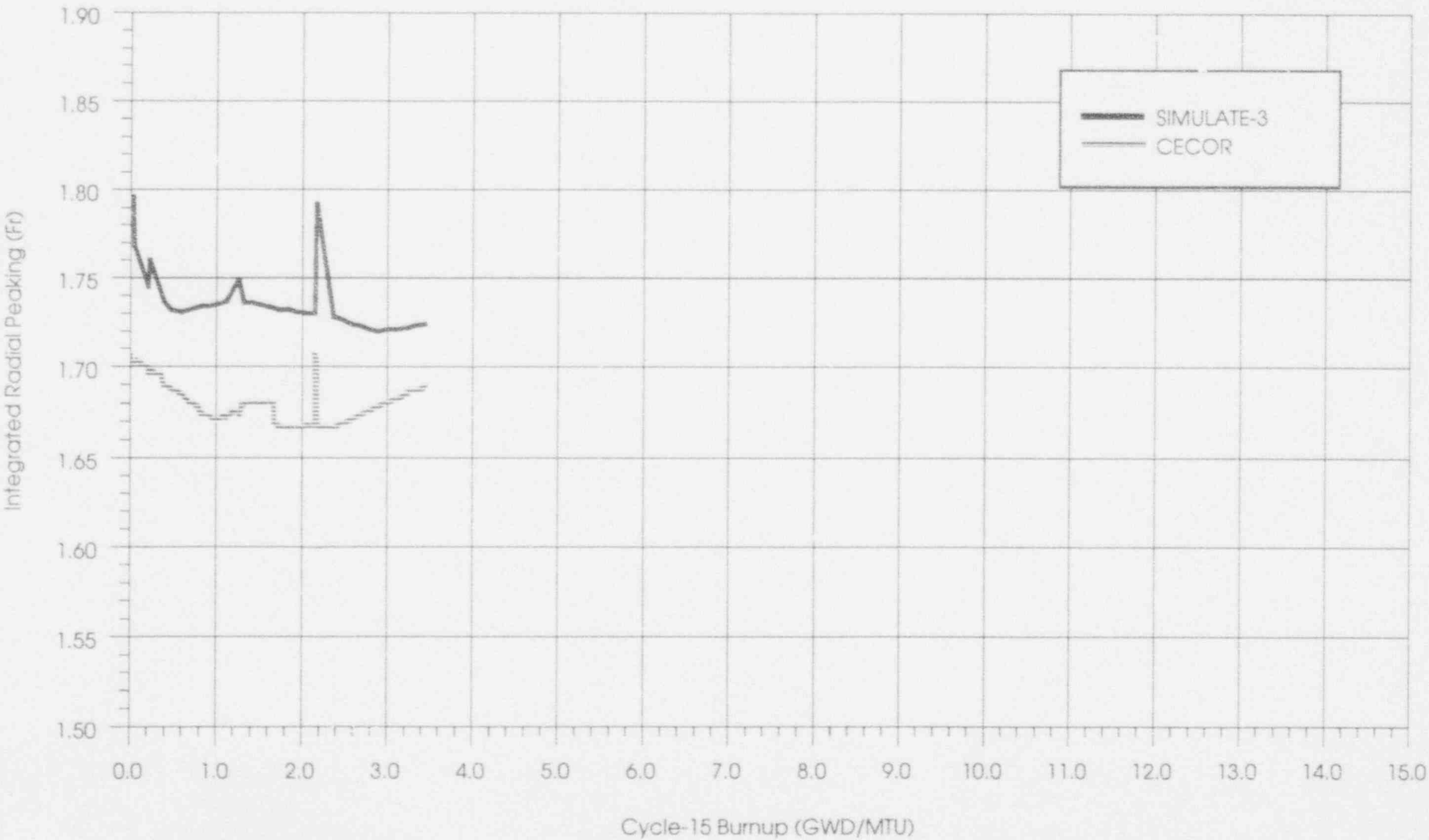


Figure 4-37

Cycle 11 Planar Radial Peaking (F_{XY}) vs. Burnup

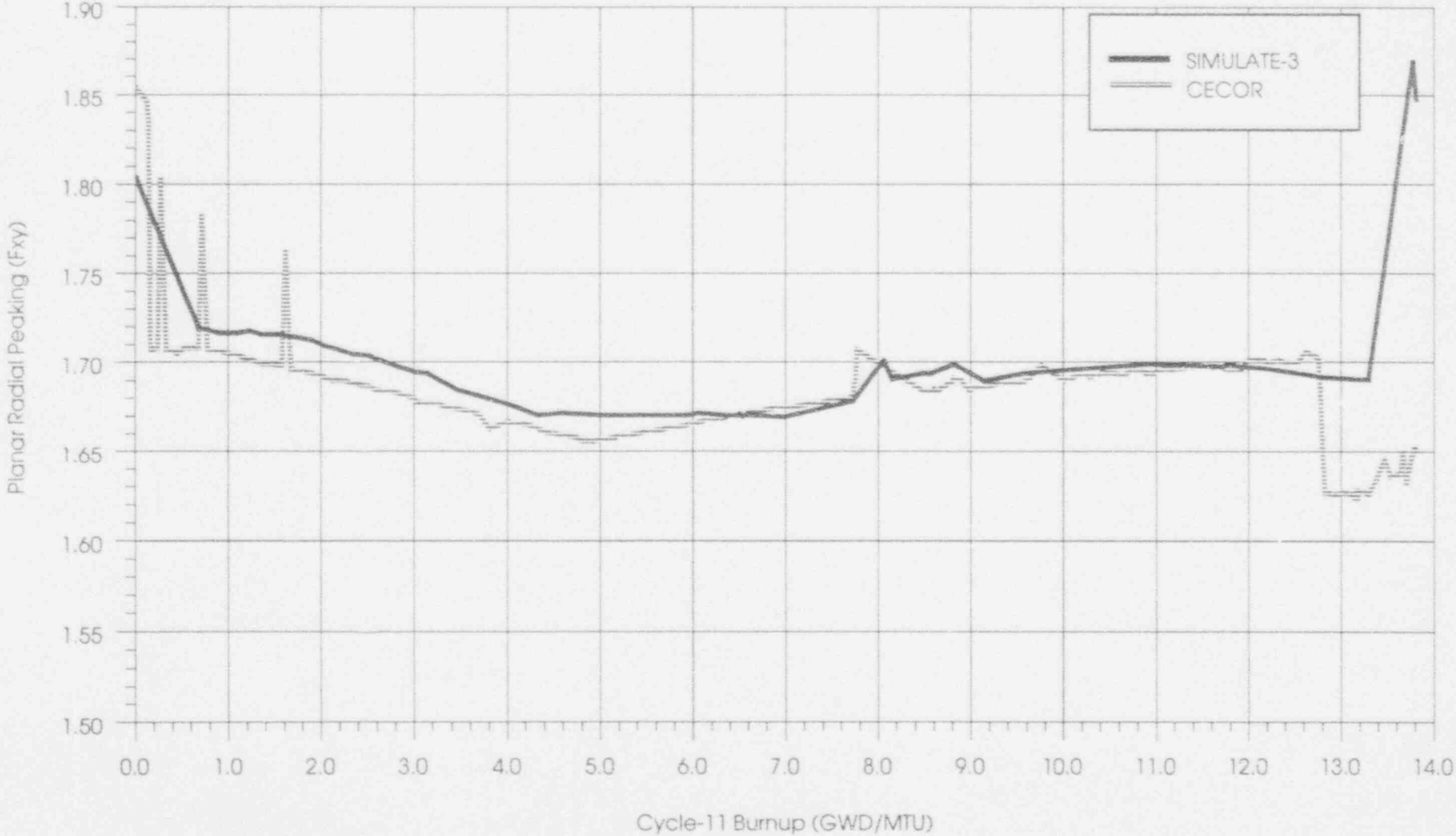


Figure 4-38

Cycle 12 Planar Radial Peaking (F_{XY}) vs. Burnup

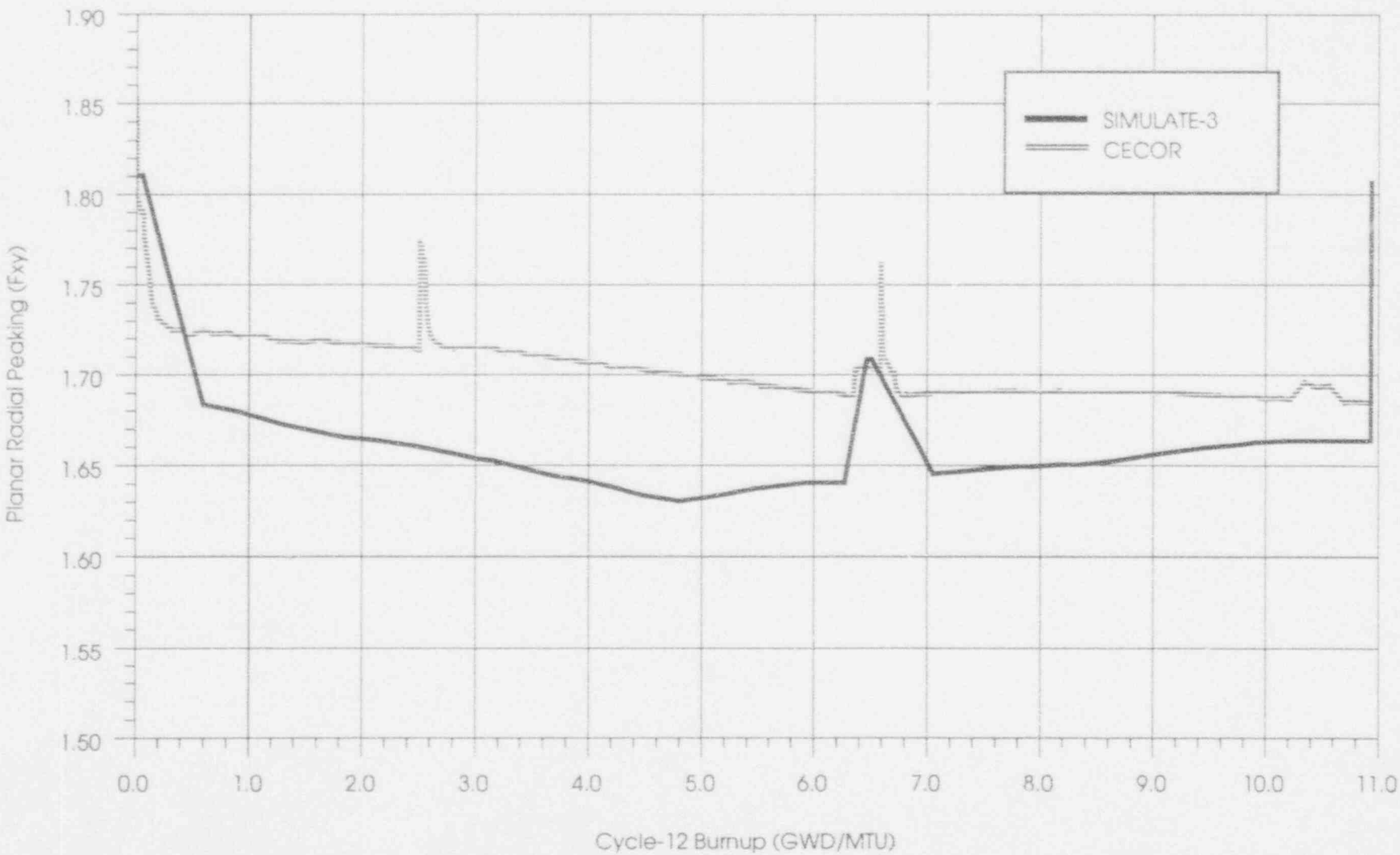


Figure 4-39

Cycle 13 Planar Radial Peaking (F_{XY}) vs. Burnup

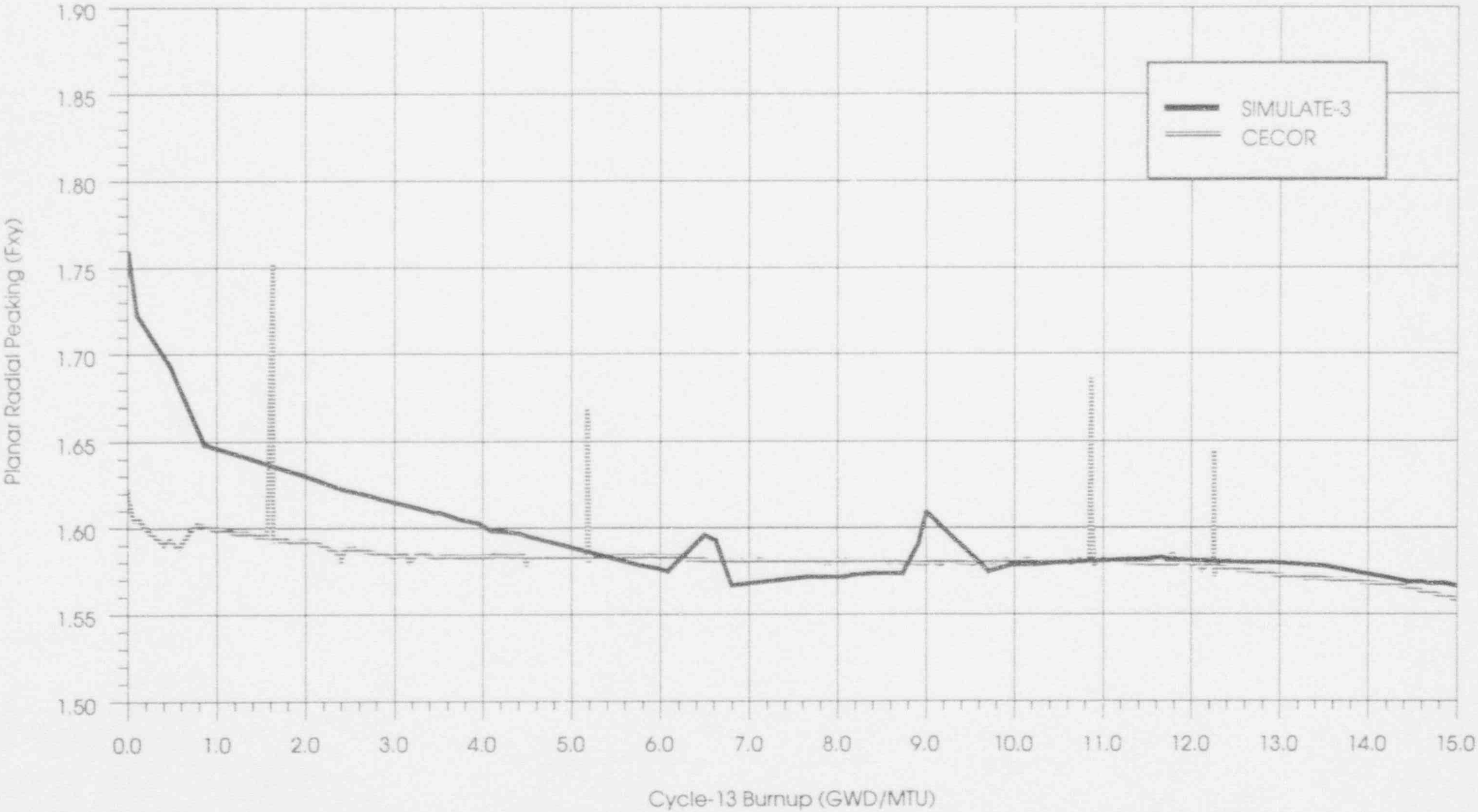


Figure 4-40

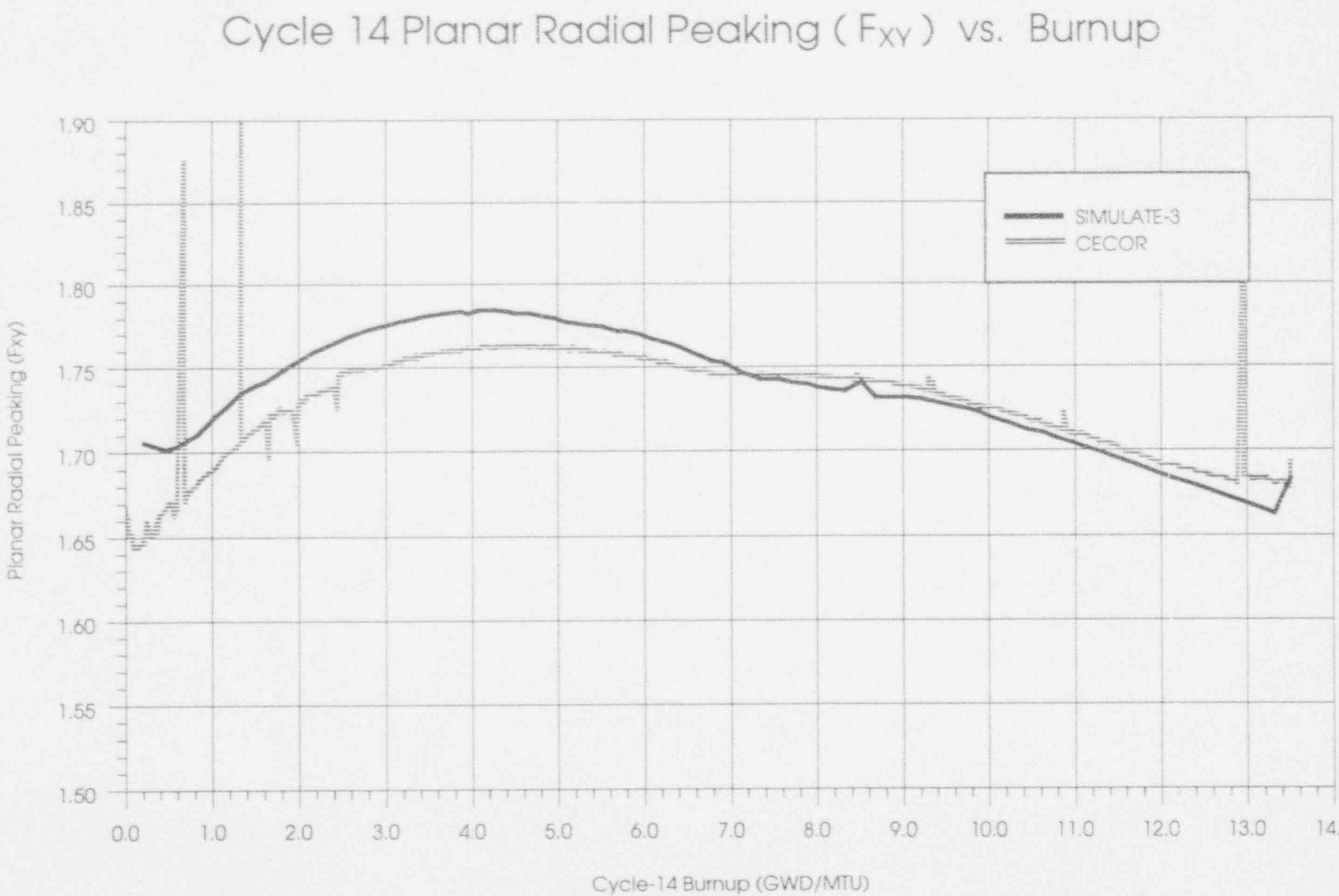

Cycle 14 Planar Radial Peaking (F_{XY}) vs. Burnup
SIMULATE-3
CECOR
Planar Radial Peaking (Fxy)
1.90
1.85
1.80
1.75
1.70
1.65
1.60
1.55
1.50
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
11.0
12.0
13.0
14.0
Cycle-14 Burnup (GWD/MTU)

Figure 4-41

Cycle 15 Planar Radial Peaking (F_{XY}) vs. Burnup

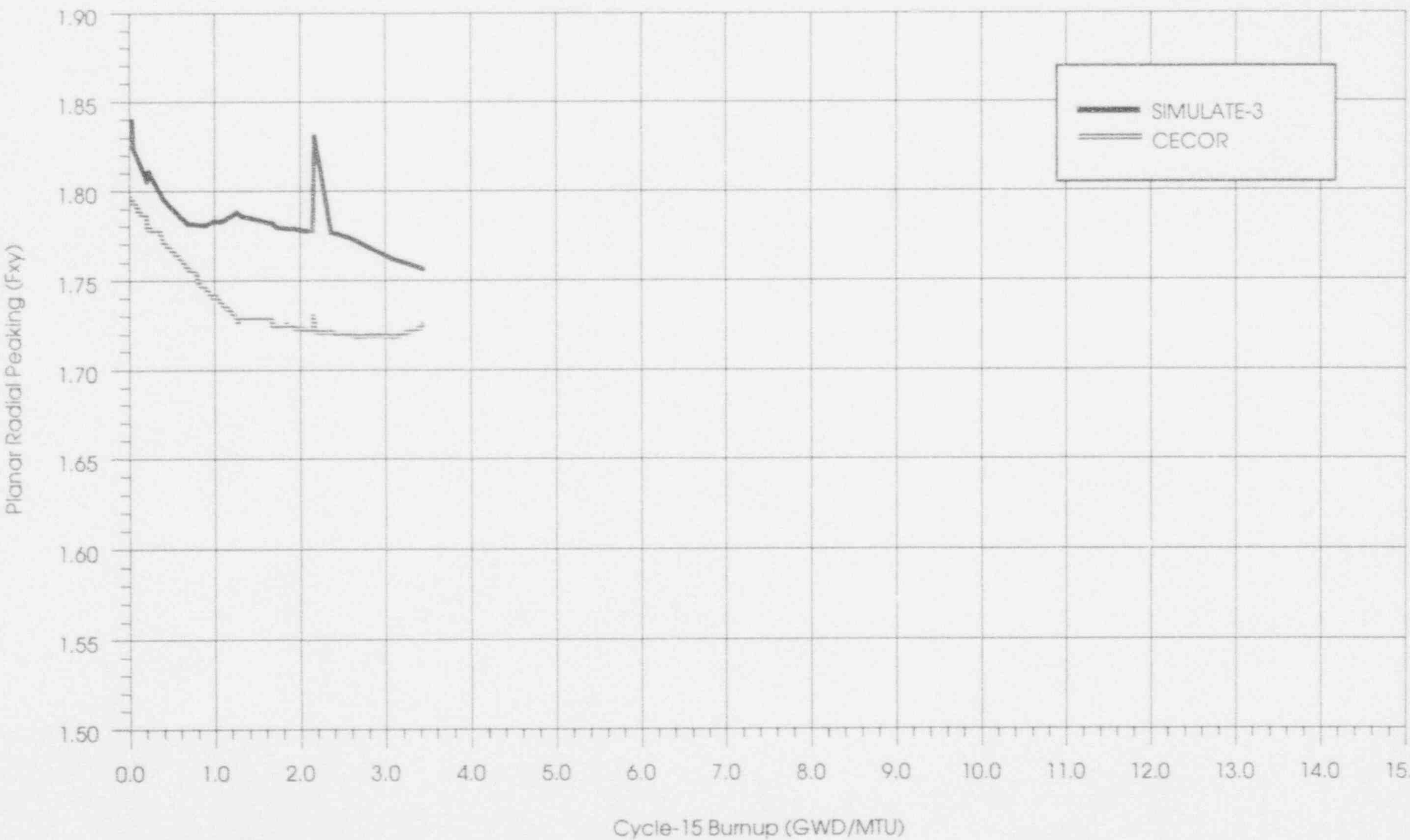


Figure 4-42

Cycle 11 3-D Peaking (F_Q) vs. Burnup
SIMULATE-3
CECOR
3-D Peaking (Fq)
2.30
2.25
2.20
2.15
2.10
2.05
2.00
1.95
1.90
1.85
1.80
1.75
1.70
1.65
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
11.0
12.0
13.0
14.0
Cycle-11 Burnup (GWD/MTU)

Figure 4-43

Cycle 12 3-D Peaking (F_Q) vs. Burnup

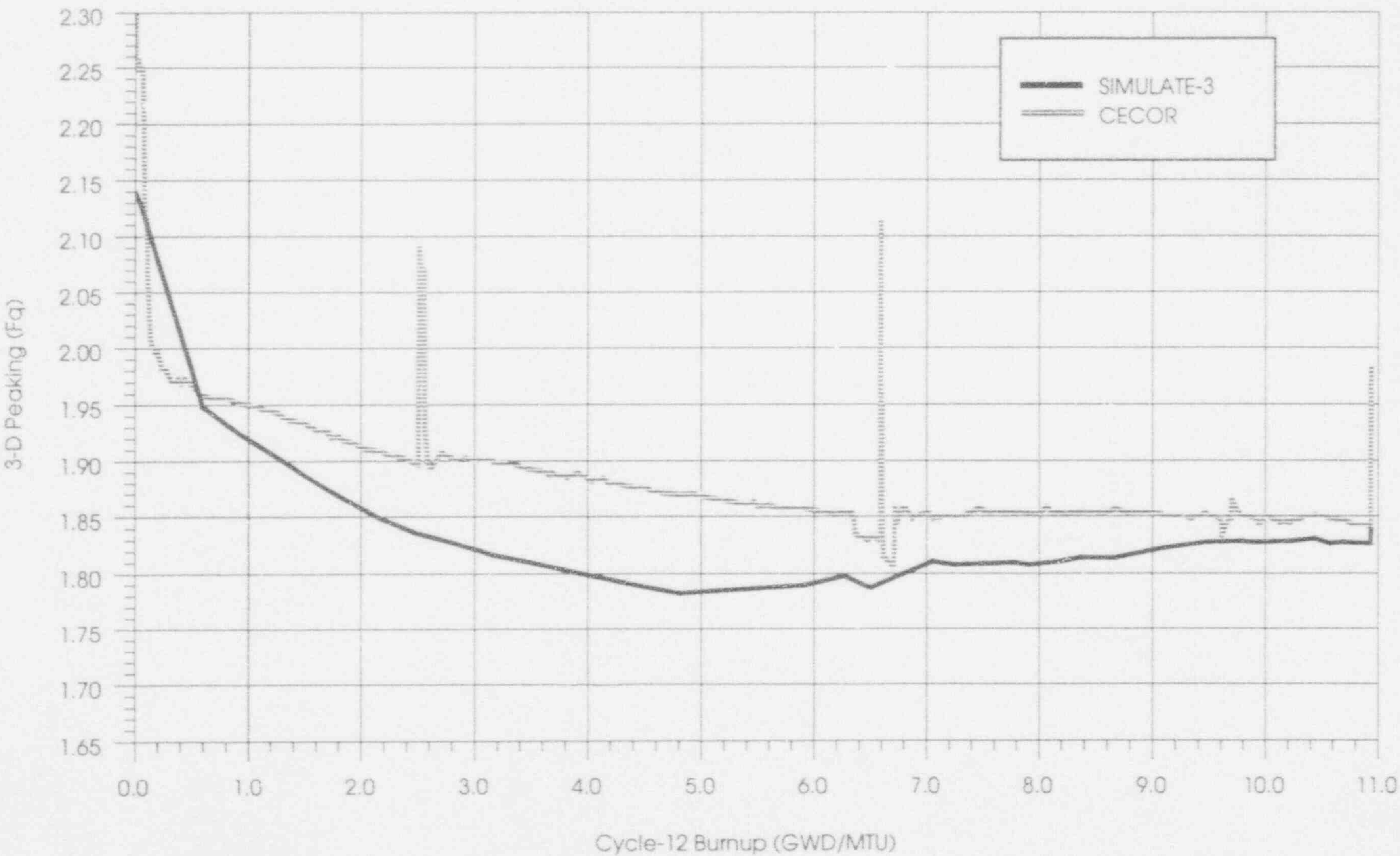
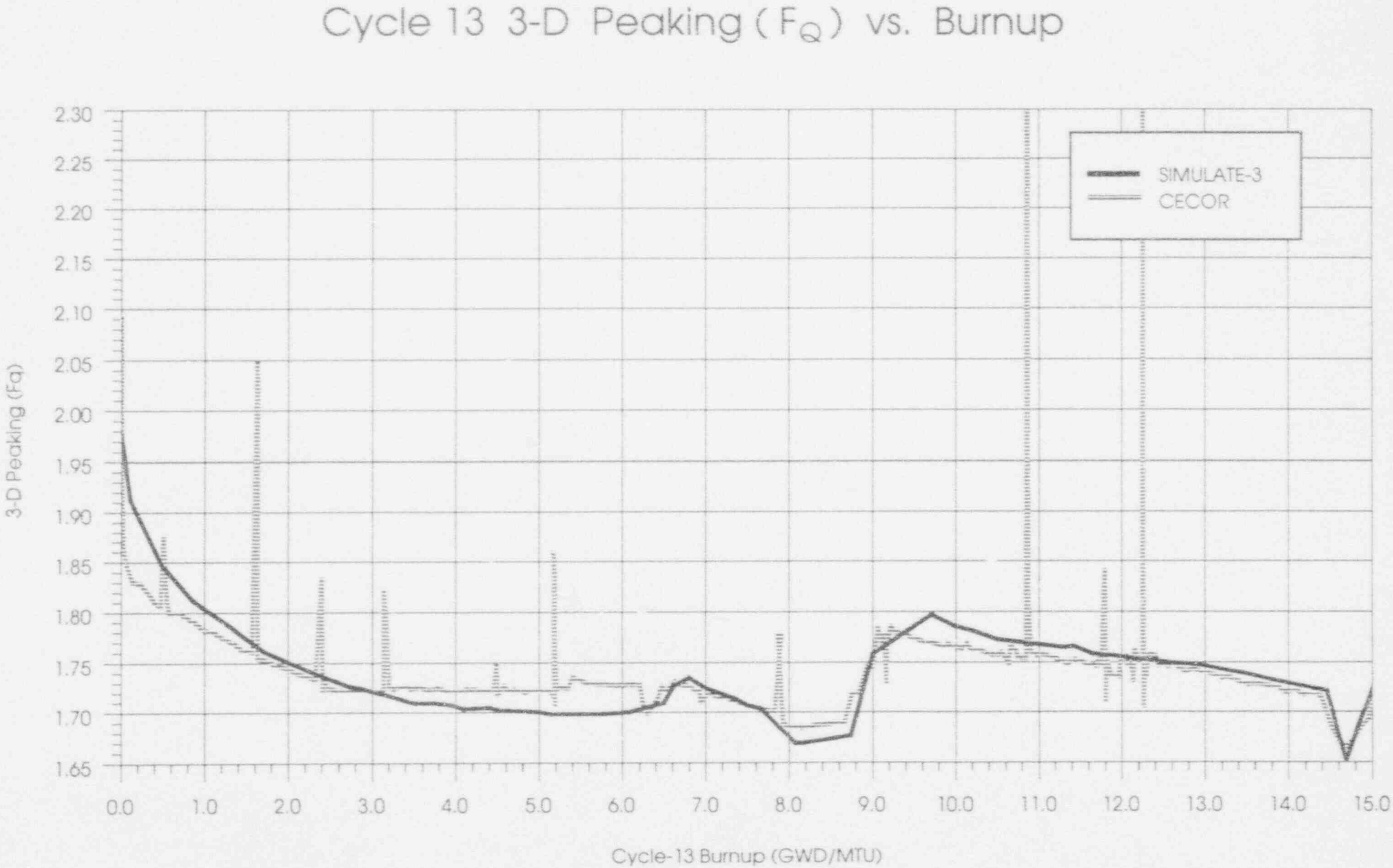


Figure 4-44



Cycle 13 3-D Peaking (F_Q) vs. Burnup
SIMULATE-3
CECOR
3-D Peaking (Fq)
2.30
2.25
2.20
2.15
2.10
2.05
2.00
1.95
1.90
1.85
1.80
1.75
1.70
1.65
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
11.0
12.0
13.0
14.0
15.0
Cycle-13 Burnup (GWD/MTU)

Figure 4-45

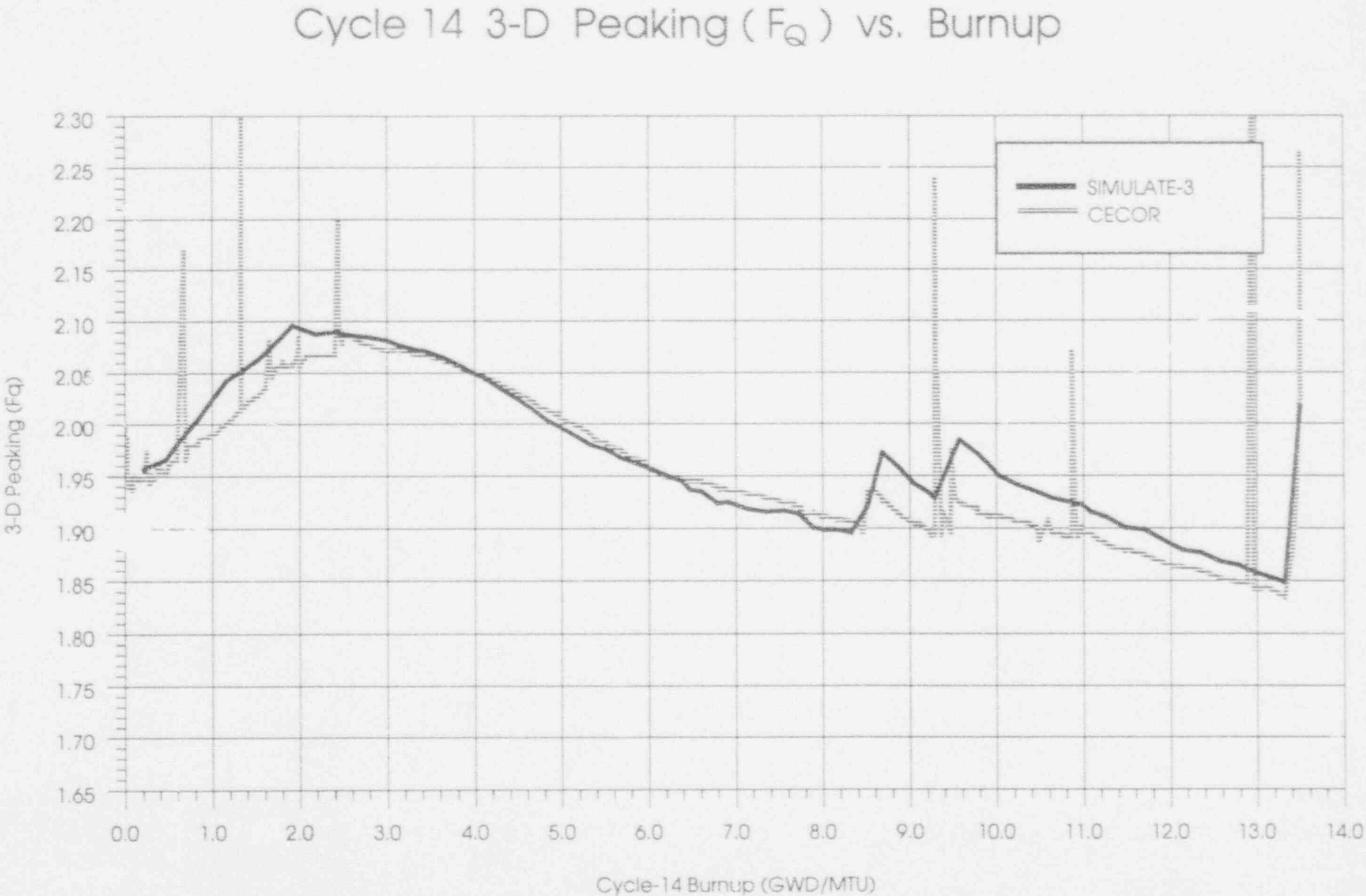

Cycle 14 3-D Peaking (F_Q) vs. Burnup
SIMULATE-3
CECOR
3-D Peaking (Fq)
2.30
2.25
2.20
2.15
2.10
2.05
2.00
1.95
1.90
1.85
1.80
1.75
1.70
1.65
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
11.0
12.0
13.0
14.0
Cycle-14 Burnup (GWD/MTU)

Figure 4-46

Cycle 15 3-D Peaking (F_Q) vs. Burnup