NUCLEAR PHYSICS METHODOLOGY FOR RELOAD DESIGN

VIRGIL C. SUMMER NUCLEAR STATION

SOUTH CAROLINA ELECTRIC & GAS COMPANY

NUCLEAR OPERATIONS DIVISION

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR PHYSICS METHODOLOGY

FOR RELOAD DESIGN

JULY 1991

SOUTH CAROLINA ELECTRIC & GAS COMPANY

NUCLEAR OPERATIONS DIVISION

JENKINSVILLE, SOUTH CAROLINA

ABSTRACT

This document summarizes the nuclear design methodology employed by SCE&G to perform reload core design analyses for VCSNS. This methodology, including all computer programs used, was obtained in its entirety from Westinghouse Electric Corporation. Calculations were performed using this methodology and the results compared to operating data from VCSNS. The quality of the comparisons demonstrates SCE&G's ability to perform reload core design for VCSNS.

TABLE OF CONTENTS

TABLE OF CONTENTS
(continued)

LIST OF TABLES

LIST OF TABLES
(continued)

LIST OF FIGURES

LIST OF FIGURES
(continued)

LIST OF FIGURES
(continued)

LIST OF FIGURES
(continued)

Figure	Page

LIST OF FIGURES
(continued)

LIST OF FIGURES
(continued)

1.0 INTRODUCTION AND CONCLUSIONS

This report briefly describes the PWR physics methods used by South Carolina Electric and Gas Company (SCE&G). It includes a summary description of the Westinghouse computer programs and methodology as applied by SCE&G to model the V. C. Summer Nuclear Station (VCSNS) core. Comparisons between predictions and operating data are provided as a demonstration of SCE&G's qualifications to use the Westinghouse methodology to perform reload design calculations for VCSNS.

1.1 OBJECTIVE

The objective of this report is to demonstrate SCE&G's capability to perform reload design analyses for VCSNS. To this end, design calculations have been performed for Cycles 3, 4, and 5 and the results are compared to actual plant operating data herein.

1.2 BACKGROUND

SCE&G has realized that in-house capability to design reload cores for VCSNS would provide the following benefits:

- A better understanding of the design, yielding more control of the decision process,
- An optimization of the design, allowing a greater involvement in planning, and
- Better quality control of the design, leading to more comprehensive evaluations of core safety.

Various physics methodologies were reviewed to determine which best satisfied SCE&G's needs. SCE&G decided to use the approach of Westinghouse, our NSSS vendor and present fuel supplier. The Westinghouse methodology offers three distinct advantages:

- A systematic physics methodology developed for and previously applied to a large number of PWR designs, including VSCNS and similar plants,

- A physics methodology previously reviewed and approved by the NRC, and
- A physics methodology that is already compatible with the other analytical design processes (e.g., thermal hydraulics and safety analyses) being used for VCSNS.

Implementation of the above decision involved entering into a technology exchange agreement with Westinghouse Electric Corp. The relevant methodology and associated computer programs of the Westinghouse Commercial Nuclear Fuel Division have been transferred to SCE&G. A description of the physics models is provided in the next chapter while the computer programs themselves are discussed in the Appendix.

1.3 SCOPE

SCE&G has performed in-house design calculations and core follow analysis for VCSNS since its initial startup. Core follow results acquired during Cycles 3, 4, and 5 provide reliable data to which predicted power distributions, predicted boron letdown curves, and fuel depletion calculations may be compared. In addition, the startup physics measurements made during the startup of each cycle provide reliable data for evaluating the physics model predictions of critical boron concentrations, control rod worths, and temperature coefficients. Detailed comparisons are presented in Section 4.

All methods employed (model development, computer programs, etc.) to generate the results contained in this report are standard licensed methods used by the Westinghouse Commercial Nuclear Fuel Division. Therefore, the calculational uncertainties (e.g., see Reference 1) associated with the methods are unchanged and their requantification is unnecessary. Similarly, the methods utilized to process measured data (e.g., see Reference 2), are also standard to Westinghouse such that measurement uncertainties do not require redetermination.

1.4 CONCLUSIONS

This report summarizes the Westinghouse methodology used by SCE&G to model the VCSNS core. Calculations were performed for Cycles 3, 4, and 5 and the

results are compared to actual operating data. Cycles 1 and 2 were also modeled to establish the appropriate burnup distributions. The results demonstrate that SCE&G understands the methodology and can apply it correctly during the performance of future reload design analyses for the V. C. Summer Nuclear Station.

"This page intentionally blank."

2.0 PHYSICS METHODOLOGY

This section briefly describes the Westinghouse methodology used by SCE&G to perform design calculations for reload cores. The major features associated with each model are discussed as well as the interaction between models. This methodology was also used to obtain the results presented in Section 4. Descriptions of the individual computer codes used are provided in the Appendix. Detailed discussions are contained in the Westinghouse documentation referenced here and in the Appendix.

Lattice physics parameters for unit assemblies and baffle-reflector cross sections are calculated with the two-dimensional multi-group transport theory code, PHOENIX-P. Fuel and clad temperatures are generated with the FIGHTH code. The core is modeled in three dimensions with the advanced nodal code, ANC, which is used to predict reactivity, power distributions, and other relevant core characteristics. In addition, the one-dimensional diffusion theory code, APOLLO, is used to calculate differential control rod worths and axial power distributions for the heat flux hot channel factor (F_Q) synthesis to establish operational limits. The cross section library as well as PHOENIX-P, nodal, and diffusion theory models are discussed in the following sections.

The models described here are representative of current Westinghouse practice. SCE&G's calculational capabilities are expected to evolve in parallel with Westinghouse's through continued implementation of the technology exchange agreement between the two organizations.

2.1 CROSS SECTION LIBRARY

The nuclear cross section library used by the PHOENIX-P computer program contains microscopic cross section data based on a 42 energy group structure which has been derived from ENDF/B-V files. The PHOENIX-P cross section library was designed to properly capture integral properties of the multigroup data during the group collapse, enabling accurate modeling of important resonance parameters, and to provide the overall accuracy of reactivity predictions required for core design. It has been developed in a manner consistent with existing Westinghouse methodologies and accumulated

experience in core design. The generation and benchmarking of the PHOENIX-P library are described in detail in Reference 3.

2.2 PHOENIX-P LATTICE MODELING

In PHOENIX-P, the fuel, discrete absorbers, and structural components within a single fuel assembly are represented in their exact lattice configuration. Homogenized two-group microscopic cross sections, discontinuity factors, and pin factors are generated as a function of burnup for input to ANC. Microscopic cross sections are generated for isotopes and materials represented explicitly in ANC. These include xenon, samarium, soluble boron, water, and burnable absorbers. Branch calculations are performed at selected burnups to obtain constants for rodded assemblies.

PHOENIX-P allows a three region cylindrical cell description for each cell within the lattice. Since most lattice cells consist of more than three subregions, material preservation principles are employed to construct a three region cell representation. The third or outer region of each cell, defined by the fuel pin pitch, has a common composition in all cells in a given lattice problem. The grids are modeled by smearing the grid material uniformly over this common outer region. Only grids in the active fuel are smeared. The following sections describe the different types of cell models.

2.2.1 FUEL CELL MODEL

The innermost region of a fuel rod cell is defined by the fuel pellet outer radius. The second region is defined by the clad outer diameter and includes the gap. For fresh fuel, the appropriate number densities are specified for the uranium isotopes and oxygen. For burned fuel, isotopic information for the depletion and decay chains modeled in PHOENIX-P is obtained from previous depletion calculations. Fuel pellets with integral fuel burnable absorber (IFBA) are not explicitly modeled with a coating on the pellet; rather, the absorber material is smeared into the clad region. PHOENIX-P corrects for the effect upon reactivity of modeling the absorber in the clad instead of on the pellet.

2.2.2 DISCRETE ABSORBER MODELS

A. BURNABLE ABSORBER RODS

Two types of discrete burnable absorber (BA) rods have been used at VCSNS: Pyrex glass and wet annular burnable absorber (WABA). The Pyrex BA is voided in the central region while the WABA contains moderator material, hence the cell representation for the two BA types is significantly different. Also, for BA's, the surface area of the absorber material must be preserved in addition to the amount of material.

For a Pyrex BA, the void, inner clad, and BA pellet material are smeared into the first region with a radius equal to the BA pellet outer radius. Region 2 comprises the BA outer clad, gap, guide tube and sleeve volumes, and materials. Note that the small volume of moderator between the BA outer clad and the guide tube is modeled as if it is outside the guide tube. Since the zircaloy guide tube material is nearly transparent to neutrons, this is a minor approximation.

In the case of WABA's, both the inner and outer surfaces of the absorber are important since a fast neutron can pass through the absorber, become thermalized in the inner region, and be absorbed. Therefore, region 1 of the cell is defined as moderator material with an outer radius equivalent to the BA pellet inner radius and region 2 as pure pellet material with an outer radius equivalent to the BA pellet outer radius. The inner clad, inner gap, outer gap, outer clad, guide tube, and sleeve materials are smeared into the moderator region to preserve materials.

B. CONTROL RODS

Control rod cells are modeled the same as Pyrex BA cells; the only distinction is the dimension of and material in the pellet region. PHOENIX-P performs resonance calculations for the Ag-In-Cd control rod material.

2.2.3 STRUCTURAL CELL MODELS

Structural cells are cells that contain neither a strong absorber nor material that is depletable. These include guide tubes, instrumentation tubes, water displacer rods, and stainless steel rods. Typically these can be represented with three or fewer regions and do not require any special neutronic considerations. Sleeves are accounted for by calculating an effective guide tube thickness that preserves the sleeve volume.

2.3 BAFFLE-REFLECTOR MODELING

A one-dimensional slab calculation is performed with PHOENIX-P to generate baffle-reflector cross sections for ANC. The model consists of a series of fuel cells approximating two fuel assemblies, assembly/baffle gap, baffle, reflector, core barrel, thermal pad (on the flats), and moderator. A set of homogenized cross sections for ANC is obtained which reflect the complex spectrum variation which exists between the fuel assemblies, baffle, and reflector.

2.4 THREE-DIMENSIONAL NODAL MODEL

The homogenized cross sections, discontinuity factors, and pin factors generated on a cycle specific basis with PHOENIX-P depletion calculations are used to model the three-dimensional core in ANC. Each fuel assembly is represented by four radial nodes. To obtain an accurate pin power recovery solution, the burnup gradient within each node is represented in ANC. A burnup gradient algorithm [illegible] node corner and surface average burnups.

[illegible]ly heterogeneous features such as axial blankets and part length burnable absorbers are explicitly represented using the variable axial mesh capability in ANC. Generally, 20 to 24 axial mesh intervals produce accurate axial power distributions. Axial zoning of burnup dependent fuel cross sections is used to account for spectrum effects induced by axial burnable absorber and fuel burnup gradients. Previous cycle burnable absorber history effects are also accounted for by using different sets of fuel cross sections.

Three-dimensional ANC calculations are used to predict core power distributions, peaking factors, critical boron concentrations, control rod worths, and reactivity coefficients. The three-dimensional model can also be collapsed to two-dimensions for certain calculations (e.g., selection of the highest worth stuck rod) where a three-dimensional representation is not necessary.

2.5 ONE-DIMENSIONAL DIFFUSION THEORY MODEL

The three-dimensional ANC model is radially homogenized to generate a one-dimensional APOLLO model. Cross sections are flux and volume weighted, and a burnup and elevation dependent radial buckling search is performed to normalize the APOLLO model to ANC. The axial mesh is redefined to comprise 40 or more axial intervals. The one-dimensional diffusion theory model is used for calculations where additional detail is desirable in the axial direction. These include generation of differential and integral control rod worth curves, determination of control rod insertion limits, and analysis of axial power distributions to establish limits on axial offset during power operation.

"This page intentionally blank."

3.0 PHYSICS MODEL APPLICATIONS

The physics methodology discussed in Section 2 was developed in order to provide reliable analytical predictions in the following four major areas:

- Core power distributions at steady state conditions,
- Axial power distribution control limits,
- Core reactivity parameters, and
- Core physics parameters for transient analysis input.

Often more than one model may be used to perform a specific analysis. The preferred model depends upon a number of considerations including the degree of accuracy desired, the specific applications, and the required resources.

The references made in this section refer to the specific models described in Section 2. SCE&G will continue to upgrade the methodology used such that it remains current with the latest approved calculational techniques being employed by Westinghouse.

3.1 CORE POWER DISTRIBUTIONS AT STEADY STATE CONDITIONS

The prediction of steady-state core power distributions is fundamental to the design, analysis, and surveillance of nuclear reactor cores. Accurate prediction of core power distributions leads to confidence in developing and optimizing core loading patterns, ensuring compliance with Technical Specification limits, and determining fuel assembly burnups and isotopic inventories.

3.1.1 POWER DISTRIBUTIONS

Global core power distributions are obtained as a function of burnup from three-dimensional ANC depletion calculations. Calculations are also performed at selected burnups for various power levels and control rod configurations. Peak rod powers and hot channel factors are generated by pin power reconstruction within ANC using rod-by-rod power distributions from

single assembly two-dimensional PHOENIX-P fine mesh spectrum calculations.

3.1.2 POWER PEAKING

Local power peaking is monitored to ensure that the peak pellet power and the total energy content within each coolant channel remain within Technical Specification and /or fuel design limits. The factors used to measure local power peaking include:

- the heat flux hot channel factor, F_Q, defin[illegible]ximum local heat flux on the surface of a fuel rod divided by th[illegible]erage fuel rod heat flux,
- the nuclear enthalpy rise hot channel factor, $F_{\Delta H}$, defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power, and
- the planer radial power peaking factor, $F_{XY}(Z)$, defined as the ratio of the peak pin power density to the average pin power density in the horizontal plane at elevation z.

For steady state conditions, these are obtained from three-dimensional ANC calculations using pin power reconstruction. For maneuvering and transient xenon conditions, a three-dimensional, one-dimensional synthesis technique (see Section 3.2) is used.

3.1.3 FUEL DEPLETION

Three-dimensional fuel depletion calculations are performed with ANC. Rod-by-rod burnup distributions are obtained from the ANC depletions; specific fuel nuclide inventories are obtained from two-dimensional single assembly PHOENIX-P depletion calculations.

3.2 AXIAL POWER DISTRIBUTION CONTROL LIMITS

The axial power distribution is primarily affected by control rod position, xenon and burnup distributions, and temperature. Axial power distribution control limits are used to ensure that thermal limits are not violated during power level

changes, control rod motion, and the resulting xenon redistributions. This is accomplished by maintaining the axial flux difference within acceptable boundaries. Axial flux difference, ΔI, is defined as the difference between the upper and lower excore detector signals, divided by the sum of these two signals.

Axial power distribution control limits are determined using Westinghouse's Relaxed Axial Offset Control (RAOC) calculational procedure (Reference 4). The RAOC calculational procedure begins by defining tentative ΔI limits which are wider than the expected LOCA limits (or, alternately, the RAOC ΔI limits from the previous cycle may be used if it is desired only to verify their acceptability). Xenon transient simulations are performed with the one-dimensional APOLLO code at various burnups and for different power levels, constrained by the tentative ΔI limits and power dependent rod insertion limits. A library of axial xenon shapes is constructed at each burnup. Next, axial power shapes are generated with APOLLO for all possible combinations of xenon shapes, power levels, and rod insertions. These axial shapes are synthesized with height dependent planer radial power distributions from three-dimensional ANC calculations. Imposition of the LOCA kw/ft limits for normal operation then defines the allowable ΔI limits (or verifies that the previous cycle's limits are acceptable) for the cycle. The axial power shapes corresponding to cases within the ΔI limits are checked against thermal hydraulic constraints from loss of flow accident simulations and the peak power and DNB limits for accident conditions.

For normal operations, more restrictive ΔI limits are developed if either the kw/ft limits or thermal hydraulic constraints are exceeded. For accident conditions, analyses are performed to verify that all design limits are met. If necessary, trip setpoints may be revised and/or the RAOC ΔI limits tightened. Therefore, the RAOC procedure provides axial power shape information which is used to verify that all design limits are met. The RAOC ΔI limits are placed in the Core Operating Limit Report and apply during plant operation.

3.3 CORE REACTIVITY PARAMETERS

The core reactivity is affected by changes in the reactor which occur during operation as the result of fuel depletion, operator actions, and abnormal or accident conditions. Reactivity coefficients quantify the rate of reactivity change

to be expected in response to changes in power, moderator or fuel temperatures, and soluble boron concentration. Reactivity defects refer to the integral of the corresponding reactivity coefficient between two reactor statepoints with all other variables remaining constant. Xenon, samarium, and control rod worths are also typically required to fully define the change in reactivity between two core configurations. In addition, neutron kinetics parameters are needed to describe the time dependent behavior of the core.

Quantification of these effects is needed for input to safety analyses as well as to provide guidance to the reactor operators and ensure compliance with Technical Specifications. Therefore, the physics models described in Section 2 are used to calculate reactivity coefficients, reactivity worths, and kinetics parameters as a function of core burnup, moderator temperature, and power level.

3.3.1 MODERATOR TEMPERATURE COEFFICIENT

The moderator temperature coefficient (MTC) is defined as the change in reactivity per degree change in moderator temperature. The effect of concomitant changes in moderator and soluble boron densities are included. The MTC is sensitive to the values of the moderator density, moderator temperature, soluble boron concentration, fuel burnup, and the presence of control rods and/or burnable absorbers which reduce the required soluble boron concentration and increase the leakage of the core. The MTC may be positive or negative depending on the magnitude of change of the individual components of this coefficient.

The MTC is calculated using the ANC core model described in Section 2.4 by varying the moderator temperature around a reference temperature. The moderator temperature coefficient is analyzed for various reactor conditions, from hot zero power (HZP) to hot full power (HFP), for various boron concentrations and control rod positions, and at various cycle burnups. The moderator temperature defect is also obtained using the ANC core model.

3.3.2 DOPPLER TEMPERATURE COEFFICIENT

The Doppler temperature coefficient is defined as the change in reactivity per degree change in effective fuel temperature. The effective fuel temperature accounts for the spatial variation in fuel temperature throughout the core. The Doppler power coefficient represents the corresponding change in reactivity per percent change in reactor power. These coefficients are primarily a consequence of the Doppler broadening of U-238 and Pu-240 resonance absorption peaks which increases the effective resonance absorption cross section of the fuel with increasing fuel temperature.

The Doppler power coefficient is normally calculated using the ANC core model by varying the reactor power level about a reference power (which in turn varies the fuel temperature) while holding the product of the power level and the enthalpy rise constant. The Doppler power coefficient is then converted to a temperature coefficient using a power/temperature relationship obtained from FIGHTH calculations. The FIGHTH code provides effective fuel temperatures, which account for spatial variations in temperature within the pellet, as a function of power level and burnup. The Doppler coefficient is analyzed at different power levels and for various cycle burnups. Doppler reactivity defects can also be obtained using the ANC model by varying the reactor power at various times in life, while holding the product of the power level and the enthalpy rise constant.

At hot zero power, the Doppler temperature coefficient may be calculated by subtracting the moderator temperature coefficient from the isothermal temperature coefficient, provided the latter has been calculated explicitly (see Section 3.3.4).

3.3.3 TOTAL POWER COEFFICIENT

The total power coefficient is defined as the change in reactivity per percent change in core power level. It represents the combined effect of moderator temperature and fuel temperature changes for an associated change in core power level.

The total power coefficient is calculated using the ANC core model by varying the core power level around a reference value while allowing the inlet temperature to change in accordance with the inlet program for the plant. The power coefficient is analyzed at different power levels and at various times in core life. The power defect is also obtained using the ANC model by varying the reactor power.

3.3.4 ISOTHERMAL TEMPERATURE COEFFICIENT

The isothermal temperature coefficient (ITC) is defined as the change in reactivity per uniform degree change in core temperature. Normally calculated only at hot zero power, the ITC is the temperature coefficient directly measured during startup physics testing. The ITC can be calculated by summing the moderator temperature coefficient and the Doppler temperature coefficient. Alternately, the ITC may be calculated explicitly using the ANC core model by varying both the moderator temperature and the fuel temperature about a uniform reference temperature.

The isothermal temperature defect (ITD) refers to the change in reactivity between hot zero power temperatures and temperatures below hot zero power. ITDs are needed as a function of temperature and burnup for various rod patterns to establish shutdown boron concentration requirements. They are calculated with the ANC model using cross sections generated with PHOENIX-P at specific temperatures between hot zero power and 68°F.

3.3.5 BORON REACTIVITY COEFFICIENT

The boron reactivity coefficient, also referred to as the differential boron worth, is defined as the change in reactivity per ppm change in the soluble boron concentration. The inverse of the boron reactivity coefficient is referred to as the inverse boron worth. It provides a means of determining the change in soluble boron concentration necessary to compensate for a given reactivity change. The magnitude of the boron reactivity coefficient depends primarily on the soluble boron concentration, the moderator temperature, control rod insertion, and the presence of burnable absorbers.

The boron reactivity coefficient is calculated using the ANC core model by perturbing the boron concentration in both directions about a reference value and computing the reactivity change. Boron worths are calculated as a function of boron concentration, power level, temperature, burnup, and control rod configuration.

3.3.6 XENON AND SAMARIUM WORTHS

The fission products Xe-135 and Sm-149 possess large thermal absorption cross sections. Knowledge of the concentrations and reactivity worths of these isotopes as well as the changes which will occur in response to plant maneuvers is crucial to reactor control. Since Xe-135 is also produced by iodine decay, it initially builds up and then decays following a reduction in power or shutdown. Sm-149 is a stable isotope produced by promethium decay. Following a reactor shutdown, its concentration increases. Upon restart it gradually returns to its equilibrium value.

Equilibrium xenon and samarium worths are calculated with the ANC core model at various power levels and core burnups. Changes in their worth and axial fluctuations in isotopic concentrations during transient operation are obtained using the ANC and/or APOLLO models.

3.3.7 CONTROL ROD WORTHS

Control rod worth refers to the reactivity difference between two control rod configurations. The total control rod worth, trip reactivity shape (i.e., the inserted rod worth versus rod position), integral and differential worths of individual banks, and worths of individual rod cluster control assemblies (e.g., stuck, ejected, and dropped rods) are determined as required for startup physics testing, plant operations, and input to safety analyses.

Control rod worths are analyzed for all normal and many abnormal control rod configurations as a function of burnup, power level, and moderator temperature. Total rod worths and the integral worths of individual rod banks and rod clusters are calculated using the ANC core model. Differential rod worths are obtained with the ANC and/or APOLLO models.

3.3.8 NEUTRON KINETICS PARAMETERS

Neutron kinetics parameters, which include delayed neutron fractions, decay constants, and the prompt neutron lifetime, are required as input to the plant reactivity computer and to various safety analyses. The parameters are also input to the Inhour equation to generate core reactivity as a function of startup rate and period. The kinetics parameters are evaluated at hot full power and hot zero power conditions for various cycle burnups and control rod configurations.

The PHOENIX-P cross section library contains delayed neutron fractions and decay constants for fissionable nuclides for each of the six delayed neutron energy groups. The core averaged delayed neutron fractions are obtained by weighting the delayed neutron fractions for each group by the regionwise fraction of fissions in each isotope and the regionwise power sharing in the core. The core average decay constants are calculated in a similar manner. The fraction of fissions in each isotope are obtained from single assembly PHOENIX-P calculations. Regionwise power sharings for various core conditions are obtained using the ANC core model. A delayed neutron importance factor (to account for spectrum differences between delayed and prompt neutrons) is used to get an effective core average delayed neutron fraction.

The prompt neutron lifetime also depends upon the core composition (fuel enrichment, burnup, absorbers, etc.). Single assembly PHOENIX-P calculations provide the neutron lifetime for the fuel in each core region. The core average value is determined through a power and volume weighting process.

3.4 CORE PHYSICS PARAMETERS FOR TRANSIENT ANALYSIS INPUT

The physics models described in Section 2 are used to generate key input parameters for various safety analyses. Reference 5 provides a detailed description of how these parameters are calculated and how they are utilized

during the safety evaluation process for a reload core. This section briefly discusses how these physics parameters are determined.

3.4.1 SHUTDOWN MARGIN

Shutdown margin is defined as the amount of negative reactivity by which the core is subcritical in a particular shutdown condition. At hot zero power conditions following a reactor trip, shutdown margins are calculated using the ANC core model. The highest worth control rod is assumed to remain fully withdrawn while xenon and soluble boron concentrations remain unchanged. Doppler and moderator defects, flux redistribution, rod insertion prior to trip, and calculational uncertainties are accounted for. Shutdown margins are evaluated following trips from hot full power and hot zero power conditions as a function of burnup.

Shutdown margins are also evaluated for temperatures below hot zero power. In this case, no credit is taken for xenon. The minimum allowed shutdown requirement at a particular time in core life is typically determined by either the steamline break or boron dilution accident analysis.

3.4.2 TRIP REACTIVITY

The trip reactivity worth is defined as the control rod worth available for insertion at the time of reactor trip. It is determined with the ANC core model. The highest worth control rod is assumed to be stuck in its fully withdrawn position and the remainder of the rods are assumed to be at the rod insertion limit. Calculations are made as a function of burnup and the most limiting (i.e., smallest) value is used.

The trip reactivity shape refers to the amount of reactivity insertion as a function of rod position following a trip. It is determined using the ANC and/or APOLLO models. The most limiting shape corresponds to the minimum rate of inserted rod worth as a function of rod position, and generally results from the most bottom skewed axial power distribution. Power distributions are calculated as a function of burnup to identify the most bottom skewed power profile.

3.4.3 CONTROL ROD MISALIGNMENT

A thermal-hydraulic analysis is required to show that the departure from nucleate boiling ratio (DNBR) criteria is not violated in the event of a misaligned control rod. Misalignment accidents include:

a. dropped control rods,
b. statically misaligned control rods, and
c. a single rod withdrawal.

For core configurations with dropped control rods, control rod worths and nuclear enthalpy rise hot channel factors ($F_{\Delta H}$) are determined using the ANC core model. Rods dropped from both the unrodded state and with control rods at the rod insertion limit are considered. This information is used to confirm that DNBR limits will not be violated during the transient following a dropped control rod event.

Statically misaligned control rods are also analyzed using the ANC core model. The maximum value of $F_{\Delta H}$ is determined as a function of power level to ensure that DNBR limits will not be violated. The extreme cases of static misalignment are represented by the single dropped rod discussed above and by a rod being fully withdrawn while the remaining rods in the same bank are at the rod insertion limit. Calculations for the latter type of core configuration also provide $F_{\Delta H}$ and pin power census data to verify that less than 5% of the rods in the core will violate the DNBR limit before the plant trips during a single rod withdrawal accident.

3.4.4 BORON DILUTION

Boron dilution refers to the inadvertent admission of unborated water into the Reactor Coolant System (RCS), diluting the boron concentration and causing a loss of shutdown margin. Limiting values of the initial boron concentration and boron worth to be used in the analysis are determined with the ANC core model for refueling, cold and hot shutdown, hot standby, startup, and at power conditions. For cold and hot shutdown and hot standby conditions, the analyses

establish shutdown boron concentrations necessary to preserve required operator action times at higher RCS boron concentrations.

3.4.5 CONTROL ROD BANK WITHDRAWAL

The withdrawal of control rod banks is analyzed to confirm that the maximum differential rod worth does not exceed the limiting values assumed in safety analyses. At full power, the control banks are withdrawn in their normal sequence. To simulate worst case rod bank withdrawal from subcritical conditions, each combination of two banks adjacent in the withdrawal sequence are moved with 100% overlap at hot zero power. Calculations are performed with the ANC and/or APOLLO models taking into account the effects of adverse axial xenon distributions

3.4.6 CONTROL ROD EJECTION

The rod ejection accident is analyzed to confirm that limits on peak clad temperature and maximum energy deposition in the fuel are not exceeded and that the reactor remains subcritical following the transient. Ejected rod worths and heat flux hot channel factors ($F_{\Delta H}$) are determined using the ANC core model. Calculations are performed for various power levels at the beginning and end of cycle with control rods inserted to the rod insertion limit. Also, trip reactivity is evaluated for a combination of two control rods unavailable: one stuck and one ejected.

3.4.7 STEAMLINE BREAK

Key input parameters to the thermal-hydraulic and transient analyses of a hypothetical steamline break are determined with the ANC core model. These include temperature and power coefficients of reactivity, control rod and soluble boron worths, location of the most limiting (from a DNBR standpoint) stuck rod, kinetics parameters, and shutdown margin. The break is conservatively assumed to occur at the end of cycle with the plant subcritical at the hot zero power temperature. After statepoint parameters (inlet temperatures, pressure, flow, and thermal power) characteristic of the time at which the maximum power occurs during the transient are determined, radial

and axial power distributions, enthalpy rise hot channel factors, and reactivity information are calculated. The assumed reactor conditions are representative of a non-uniform coolant inlet temperature distribution with the most limiting control rod stuck out of the core and all other rods fully inserted. This information is then used to perform a thermal-hydraulic evaluation of DNBR.

4.0 PHYSICS MODEL VERIFICATION

Core physics model verification typically includes comparisons of predictions to plant startup and operating data. The V. C. Summer Nuclear Station (VCSNS) is currently in its sixth cycle of operation. In this section, predictions made using the physics methodology described in Section 2 are compared to zero power physics test measurements and at power operating data accumulated over the past three cycles. As stated in Section 1, the methods employed to generate the predictions reported in this section are standard licensed methods used by Westinghouse's Commercial Nuclear Fuel Division. The comparisons reported herein provide additional verification of the predictive capabilities of this methodology; however, their primary purpose is to demonstrate SCE&G's ability to perform design calculations for VCSNS.

4.1 CYCLE DESCRIPTIONS

Cycle 3 of VCSNS began operation on December 14, 1985, and shutdown on March 6, 1987, after 411 effective full power days (EFPD) corresponding to a cycle burnup of 15792 MWD/MTU. The core loading pattern for Cycle 3, including the locations and number of part length pyrex burnable absorbers and the locations of control rod banks, is shown in Figure 4.1-1. A quarter core representation is used since the core loading is symmetric. Four Vantage-5 Demonstration Assemblies containing Integral Fuel Burnable Absorbers (IFBAs), initially loaded in Cycle 2, were returned to the core.

Cycle 4 of VCSNS began operation on June 6, 1987, and shutdown on September 16, 1988, after 424 EFPD corresponding to a cycle burnup of 16264 MWD/MTU. The core loading pattern for Cycle 4, including the locations and number of wet annular burnable absorbers (WABAs), is shown in Figure 4.1-2. The four Vantage-5 Demonstration Assemblies were also returned to the core for Cycle 4.

Cycle 5 of VCSNS began operation on December 26, 1988, and shutdown on March 23, 1990, after 346 EFPD corresponding to a cycle burnup of 13670 MWD/MTU. The core loading pattern for Cycle 5, including the locations and number of IFBAs, is shown in Figure 4.1-3. Fuel batch characteristics for Cycles 3, 4, and 5 are summarized in Table 4.1-1.

4.2 ZERO POWER PHYSICS TESTS

After each refueling at VCSNS, startup physics tests are conducted to verify that the nuclear characteristics of the core are consistent with design predictions. While the reactor is maintained at hot zero power (HZP) conditions, the following physics parameters are measured:

- Critical boron concentrations,
- Isothermal temperature coefficient,
- Control rod worths, and
- Differential boron worth.

Table 4.2-1 contains the zero power physics test review criteria, which represent the maximum expected deviation between predicted and measured values for each parameter.

The following sections briefly describe the measurement and calculational techniques and summarize the results of the zero power physics tests for Cycles 3, 4, and 5. Small changes in core reactivity were measured by feeding the signal from the spare power range neutron detector into a reactivity computer which solves the point kinetics equation. The computer output was plotted on a strip chart recorder. All predictions were made with the three-dimensional ANC model described in Section 2.4.

4.2.1 CRITICAL BORON CONCENTRATIONS

Critical boron concentrations were measured by acid-base titration of reactor coolant samples taken under equilibrium conditions. Samples were taken with all rods out (ARO), that is, fully withdrawn from the core, and with Bank B, the reference bank (see Section 4.2.3), fully inserted. Critical boron searches were performed with the three-dimensional ANC model for these core configurations to obtain the predicted concentrations. The measured and predicted critical boron concentrations are compared in Table 4.2-2. All differences are within the ± 50 ppm review criteria.

4.2.2 ISOTHERMAL TEMPERATURE COEFFICIENTS

Isothermal temperature coefficients (ITCs) were measured by making small changes in the reactor coolant system temperature and determining the corresponding change in reactivity with the reactivity computer. ITCs were predicted by uniformly varying the core temperature by $\pm 5°F$ about the HZP temperature in the ANC model. The moderator temperature is varied directly; Doppler effects on reactivity are determined using fitting coefficients obtained from FIGHTH calculations. The measured and predicted ITCs are compared in Table 4.2-3. All differences are well within the review criteria of ± 3 pcm/°F.

4.2.3 CONTROL ROD WORTHS

Control rod worths were measured by the Rod Swap Technique. First, the worth of the reference bank (the bank of highest worth) was measured by boron dilution. Stepwise bank insertion was used to maintain criticality and differential worths were obtained from the reactivity computer response. The differential worths were summed to provide the integral worth of the reference bank. Then, maintaining the boron concentration at a constant value, critical configurations were established with each remaining bank fully inserted and the reference bank partially withdrawn. The integral worth of each inserted bank was determined from the critical position of the reference bank after the exchange by applying analytical corrections to account for the effect of the inserted bank on the partial integral worth of the reference bank. This procedure is described in detail in Reference 6.

The ANC model was used to predict the individual control rod bank worths as well as to generate the corrections used to infer the measured worths. The measured and predicted worths are compared in Table 4.2-4; all differences are within the review criteria listed in Ta. 4.2-1. Measured and predicted reference bank integral rod worth shapes are compared in Figures 4.2-1 through 4.2-3.

4.2.4 DIFFERENTIAL BORON WORTHS

Measured differential boron worths were obtained by dividing the measured reference bank worth (see Section 4.2.3) by the difference between the critical boron concentrations measured with all rods out and with the reference bank inserted. The differential boron worth does not change significantly over this range of boron concentration. Boron worths were predicted by varying the boron concentration by ± 25 ppm about the HZP all rods out critical boron concentration in the ANC model. The measured and predicted boron worths are compared in Table 4.2-5. All differences are well within the $\pm 15\%$ review criteria.

4.3 POWER OPERATION

In support of VCSNS Technical Specification requirements, the core power distribution is measured at least once every 31 EFPD using the incore instrumentation system. Neutron flux measurements made by movable incore fission chambers are combined with analytically determined power to reaction rate ratios using the computer program INCORE (Reference 2) to infer, i.e., "measure," a three-dimensional power distribution. The power to reaction rate ratios are generated with the three-dimensional ANC model using cross sections derived from PHOENIX-P.

In this section, measured data obtained from INCORE is compared to predictions made with the three-dimensional ANC Model. Included are:

- Power peaking factors, F_Q and $F_{\Delta H}$,
- Average assembly radial power distributions,
- Core average axial power distributions, and
- Axial offset.

Also, measured and predicted boron letdown curves are compared. Boron letdown refers to the reduction of the all rods out hot full power critical boron concentration as a function of core burnup.

4.3.1 BORON LETDOWN CURVES

Reactor coolant system boron concentrations are measured daily regardless of power level or control rod bank insertion. Critical boron concentrations measured at or very close to hot full power all rods out equilibrium xenon and samarium conditions are compared to the predicted boron letdown curves for Cycles 3, 4, and 5 in Figures 4.3-1 through 4.3.3. The predicted curves were obtained from design depletions with the three-dimensional ANC model.

Tables 4.3-1 through 4.3-3 compare measured and predicted critical boron concentrations at the time of INCORE power distribution measurements. The measured concentrations were corrected to hot full power all rods out equilibrium xenon and samarium conditions in accordance with VCSNS surveillance procedures. The predicted concentrations were obtained by performing critical boron searches with the ANC model at the specific burnups of the measurements. The mean difference between measured and predicted critical boron concentrations for all three cycles is 12 ppm with a standard deviation of 19 ppm.

4.3.2 POWER PEAKING FACTORS

The nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and the heat flux hot channel factor (F_Q) were measured using the INCORE code, as discussed above. Predicted peaking factors were obtained from three-dimensional ANC calculations performed for core conditions similar to those at the time of the measurements. Power peaking factors measured during Cycles 3, 4, and 5 are compared to predicted values in Figures 4.3-4 through 4.3-9 and in Tables 4.3-4 through 4.3-6. For $F_{\Delta H}$, the mean difference between the measured and predicted values for the three cycles is 0.58% with a standard deviation of 1.11%; for F_Q the mean difference is 2.07% with a standard deviation of 1...%. Regarding the F_Q comparisons, it is noted that spacer grid effects are inherent in the measured values but the grids are not explicitly modeled in ANC. The magnitude of this effect can be seen from Figures 4.3-28 through 4.3-45.

4.3.3 RADIAL POWER DISTRIBUTIONS

Core power distributions were measured with the INCORE code, as discussed above. The measured power distributions are typically referred to as flux maps. INCORE also produces predicted power distributions at the burnup of the flux map by interpolating between power distributions generated using the three-dimensional ANC model at specific burnups during a depletion calculation. Since the core is loaded symmetrically, ANC depletion calculations are performed assuming quarter-core rotational symmetry. The predicted power distributions are expanded to full core for comparison to the measured distributions.

Figures 4.3-10 through 4.3-27 compare measured and predicted assembly relative power distributions at selected burnups spanning Cycles 3, 4, and 5. All comparisons are for the hot full power all rods out condition since this is the normal mode of operation at VCSNS. The mean absolute difference between measured and predicted assembly relative powers is less than .014 and the standard deviation is less than .017 for these comparisons.

4.3.4 AXIAL POWER DISTRIBUTIONS AND AXIAL OFFSETS

Measured core average axial power distributions from each of the flux maps discussed in the previous section are compared to predicted axial distributions in Figures 4.3-28 through 4.3-45. The predicted distributions were obtained from three-dimensional ANC calculations performed for core conditions similar to those at the time of the flux maps. Note that since the grid straps are not modeled explicitly in the ANC model, no depressions are seen at the grid locations in the predicted distributions. This difference coupled with the normalization of both measured and predicted axial power distributions to unity causes the measured relative power to appear slightly higher between grid locations.

Axial offset refers to the percent difference between the relative power in the top half of the core and that in the bottom half of the core. Axial offsets measured using the INCORE code are compared to predicted values from ANC calculations for core conditions similar to those at the time of the

measurements in Tables 4.3-7 through 4.3-9. The mean difference between measured and predicted values for Cycles 3, 4, and 5 is 0.7% with a standard deviation of 0.9%.

4.4 SUMMARY

In this section, predictions made using Westinghouse's reload core design methodology are compared to zero power physics test measurements and at power operating data from Cycles 3, 4, and 5 of VCSNS. In all cases, the predictions agree very well with the measurements. All startup test predictions are within the review criteria listed in Table 4.2-1. Predicted critical boron concentrations at power are well within 50 ppm of the measured values, and the predicted power distributions are quite close to the measured, as evidenced by Figures 4.3-10 through 4.3-45. The excellent agreement between the predictions and the measurements reported here demonstrates SCE&G's capability to apply the Westinghouse licensed methodology to reload core design for VCSNS.

"This page intentionally blank."

TABLE 4.1-1

V.C. SUMMER NUCLEAR STATION
FUEL SPECIFICATION

Cycle	Batch	Number of Assemblies	Initial Enrichment w/o U-235	BOC Burnup MWD/MTU
3	2	5	2.61	20549
	3	40	3.11	20632
	4	40	3.44	10876
	4A(a)	4	3.42	11127
	5	68	3.84	0
4	4	21	3.44	25833
	4A(a)	4	3.42	29743
	5	76	3.84	16395
	6	56	3.60	0
5	3	1(b)	3.11	24855
	4	4	3.44	18877
	5	40	3.84	26083
	6	48	3.60	18909
	7A(a)	36	3.79	0
	7B(a)	28	4.20	0

(a) Vantage-5 Fuel Design

(b) Discharged at EOC2

TABLE 4.2-1

V.C. SUMMER NUCLEAR STATION
HZP PHYSICS TEST REVIEW CRITERIA

Parameter	Review Criteria
Critical Boron Concentrations	± 50 ppm
Isothermal Temperature Coefficient	± 3 pcm/°F
Control Rod Bank Worths:	
Reference Bank Worth	± 10%
"Swap" Worths	± 15% or 100 pcm, whichever is greater
Differential Boron Worth	± 15%

TABLE 4.2-2

V.C. SUMMER NUCLEAR STATION
CRITICAL BORON CONCENTRATION COMPARISON
BETWEEN MEASUREMENT AND PREDICTION
FOR CYCLES 3, 4 AND 5

Cycle	Bank Configuration	Critical Boron Concentration (ppm)		
		M	P	(M-P)
3	ARO	1840	1837	3
	BANK B in	1635	1648	-13
4	ARO	1831	1807	24
	BANK B in	1650	1628	22
5	ARO	1969	2013	-44
	BANK B in	1821	1868	-47

TABLE 4.2-3

V.C. SUMMER NUCLEAR STATION
ISOTHERMAL TEMPERATURE COFFICIENT COMPARISON
BETWEEN MEASUREMENT AND PREDICTION
FOR CYCLES 3, 4 AND 5

Cycle	Bank Configuration	ITC (pcm/°F) M	P	(M-P)
3	ARO	-1.95	-1.09	-0.86
4	ARO	-2.89	-2.23	-0.66
5	ARO	+3.15	+2.35	0.80

TABLE 4.2-4

V.C. SUMMER NUCLEAR STATION
CONTROL ROD WORTH COMPARISON
BETWEEN MEASUREMENT AND PREDICTION
FOR CYCLES 3, 4 AND 5

Cycle	Bank Configuration	Control Rod Worth (pcm) M [a]	P	$(\frac{M-P}{P})\%$
3	Bank D	878.8	967.0	-9.12
	Bank C	738.4	854.0	-13.54
	Bank B [b]	1393.5	1398.0	-0.32
	Bank A	605.0	587.0	3.07
	Bank SB	780.2	902.0	-13.50
	Bank SA	1204.4	1184.0	1.72
4	Bank D	1089.6	1070.5	1.78
	Bank C	845.9	816.5	3.60
	Bank B [b]	1353.9	1324.2	2.24
	Bank A	443.7	425.2	4.35
	Bank SB	882.1	869.2	1.48
	Bank SA	1134.1	1124.4	0.86
5	Bank D	932.0	1003.7	-7.14
	Bank C	654.2	727.4	-10.06
	Bank B [b]	1077.9	1125.6	-4.24
	Bank A	626.5	621.9	0.74
	Bank SB	811.4	910.9	-10.92
	Bank SA	956.5	991.7	-3.55

(a) Measured Rod Worths were determined using "Rod Swap" Methodology
(b) Reference Bank

TABLE 4.2-5

V.C. SUMMER NUCLEAR STATION
HZP DIFFERENTIAL BORON WORTH COMPARISON
BETWEEN MEASUREMENT AND PREDICTION
FOR CYCLES 3, 4 AND 5

Cycle	Bank Configuration	Differential Boron Worth (pcm/ppm) M	P	$\left(\frac{M-P}{P}\right)\%$
3	Average Over BANK B Insertion	6.80	7.40	-8.11
4	Average Over BANK B Insertion	7.48	7.40	1.08
5	Average Over BANK B Insertion	7.28	7.76	-6.19

TABLE 4.3-1

V.C. SUMMER NUCLEAR STATION CYCLE 3
BORON LETDOWN COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	Critical Boron Concentration (ppm) M	P	(M-P)
617	1202	1233	-31
1194	1190	1197	-7
1839	1155	1162	-7
2341	1144	1132	12
2860	1105	1102	3
3365	1081	1071	10
3920	1045	1034	11
4758	1006	979	27
4988	989	965	24
5558	952	923	29
6345	909	870	39
7111	850	815	35
7606	805	779	26
8126	763	741	22
8689	713	696	17
9389	675	646	29
9646	637	626	11
10624	568	548	20
11123	520	505	15
12435	410	396	14
12608	403	382	21
13581	312	296	16
14057	268	255	13
14867	198	183	15

TABLE 4.3-2

V.C. SUMMER NUCLEAR STATION CYCLE 4
BORON LETDOWN COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	Critical Boron Concentration (ppm) M	P	(M-P)
397	1176	1211	-35
1232	1161	1152	9
2007	1099	1108	-9
3164	1060	1034	26
4154	984	968	16
5106	916	900	16
6398	814	803	11
7548	733	714	19
8608	663	631	32
9783	561	537	24
10850	468	447	21
12192	352	329	23
12784	301	277	24
13818	193	183	10
14970	96	77	19

TABLE 4.3-3

V.C. SUMMER NUCLEAR STATION CYCLE 5
BORON LETDOWN COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	Critical Boron Concentration (ppm) M	P	(M-P)
423	1440	1480	-40
2201	1356	1384	-28
3260	1261	1303	-42
4090	1224	1230	-6
5064	1135	1136	-1
6174	1031	1021	10
7438	903	887	16
8168	828	810	18
9354	709	687	22
10539	588	565	23
11647	488	453	35
12793	373	338	35

TABLE 4.3-4

V.C. SUMMER NUCLEAR STATION CYCLE 3
POWER PEAKING FACTOR COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	$F_{\Delta H}$ (Max) M	$F_{\Delta H}$ (Max) P	$F_{\Delta H}$ (Max) $(\frac{M-P}{P})\%$	F_Q (Max) M	F_Q (Max) P	F_Q (Max) $(\frac{M-P}{P})\%$
617	1.403	1.401	0.14	1.732	1.719	0.76
1194	1.403	1.395	0.57	1.712	1.709	0.18
1839	1.397	1.392	0.36	1.701	1.685	0.95
2341	1.393	1.393	0.00	1.687	1.672	0.90
2860	1.387	1.391	-0.29	1.675	1.658	1.03
3365	1.384	1.391	-0.50	1.669	1.648	1.27
3920	1.382	1.390	-0.58	1.668	1.642	1.58
4758	1.381	1.388	-0.50	1.645	1.637	0.49
4988	1.382	1.385	-0.22	1.640	1.628	0.74
5558	1.380	1.385	-0.36	1.640	1.631	0.55
6345	1.379	1.381	-0.14	1.629	1.620	0.56
7111	1.376	1.380	-0.29	1.614	1.615	-0.06
7606	1.376	1.379	-0.22	1.628	1.615	0.80
8126	1.379	1.378	0.07	1.629	1.616	0.80
8689	1.373	1.377	-0.29	1.638	1.618	1.24
9389	1.377	1.376	0.07	1.632	1.625	0.43
9646	1.377	1.376	0.07	1.635	1.630	0.31
10624	1.388	1.379	0.65	1.635	1.624	0.68
11123	1.384	1.380	0.29	1.636	1.619	1.05
12435	1.388	1.384	0.29	1.636	1.624	0.74
12608	1.390	1.385	0.36	1.658	1.625	2.03
13581	1.389	1.382	0.51	1.618	1.619	-0.06
14057	1.383	1.381	0.14	1.649	1.616	2.04
14867	1.384	1.379	0.36	1.632	1.606	1.62

TABLE 4.3-5

V.C. SUMMER NUCLEAR STATION CYCLE 4
POWER PEAKING FACTOR COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	$F_{\Delta H}$ (Max)			F_Q (Max)		
	M	P	$\left(\frac{M-P}{P}\right)\%$	M	P	$\left(\frac{M-P}{P}\right)\%$
397	1.388	1.394	-0.43	1.566	1.494	4.82
1232	1.382	1.387	-0.36	1.580	1.508	4.77
2007	1.385	1.381	0.29	1.589	1.514	4.95
3164	1.392	1.388	0.29	1.603	1.542	3.96
4154	1.401	1.393	0.57	1.607	1.559	3.08
5106	1.406	1.396	0.72	1.622	1.572	3.18
6398	1.411	1.404	0.50	1.632	1.588	2.77
7548	1.414	1.408	0.43	1.644	1.604	2.49
8608	1.419	1.413	0.42	1.637	1.610	1.68
9783	1.410	1.413	-0.21	1.654	1.609	2.80
10850	1.416	1.414	0.14	1.646	1.602	2.75
12192	1.407	1.409	-0.14	1.625	1.579	2.91
12784	1.408	1.406	0.14	1.627	1.568	3.76
13818	1.398	1.398	0.00	1.609	1.556	3.41
14970	1.388	1.388	0.00	1.586	1.543	2.79

TABLE 4.3-6

V.C. SUMMER NUCLEAR STATION CYCLE 5
POWER PEAKING FACTOR COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	$F_{\Delta H}$ (Max)			F_Q (Max)		
	M	P	$(\frac{M-P}{P})$%	M	P	$(\frac{M-P}{P})$%
423	1.464	1.438	1.81	1.926	1.863	3.38
2201	1.494	1.456	2.61	1.887	1.818	3.80
3260	1.491	1.447	3.04	1.847	1.769	4.41
4090	1.496	1.436	4.18	1.841	1.743	5.62
5064	1.470	1.421	3.45	1.771	1.730	2.37
6174	1.459	1.406	3.77	1.759	1.716	2.51
7438	1.419	1.393	1.87	1.735	1.692	2.54
8168	1.411	1.387	1.73	1.700	1.675	1.49
9354	1.395	1.377	1.31	1.685	1.652	2.00
10539	1.381	1.366	1.10	1.703	1.636	4.10
11647	1.375	1.362	0.95	1.653	1.628	1.54
12793	1.373	1.358	1.10	1.641	1.624	1.05

TABLE 4.3-7

V.C. SUMMER NUCLEAR STATION CYCLE 3
AXIAL OFFSET COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	Axial Offset (%) M	P	(M-P)
617	-0.43	-0.88	0.45
1194	-0.13	-1.51	1.38
1839	-1.22	-2.06	0.84
2341	-1.57	-2.45	0.88
2860	-1.27	-2.73	1.46
3365	-1.31	-3.06	1.75
3920	-2.59	-3.39	0.80
4758	-2.15	-3.87	1.72
4988	-2.27	-3.86	1.59
5558	-3.10	-4.18	1.08
6345	-2.42	-4.27	1.85
7111	-2.37	-4.49	2.12
7606	-3.24	-4.52	1.28
8126	-3.20	-4.54	1.34
8689	-3.92	-4.52	0.60
9389	-3.29	-4.39	1.10
9646	-3.57	-4.22	0.65
10624	-3.12	-3.82	0.70
11123	-2.98	-3.57	0.59
12435	-2.81	-2.91	0.10
12608	-4.10	-2.79	-1.31
13581	-2.41	-2.52	0.11
14057	-3.40	-2.41	-0.99
14867	-2.97	-2.19	-0.78

TABLE 4.3-8

V.C. SUMMER NUCLEAR STATION CYCLE 4
AXIAL OFFSET COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	Axial Offset (%) M	P	(M-P)
397	2.30	-0.17	2.47
1232	0.72	-1.22	1.94
2007	-0.59	-1.90	1.31
3164	-2.29	-2.79	0.50
4154	-2.56	-3.31	0.75
5106	-2.63	-3.66	1.03
6398	-3.41	-3.93	0.52
7548	-3.67	-4.19	0.52
8608	-3.31	-4.22	0.91
9783	-4.05	-3.91	-0.14
10850	-3.56	-3.42	-0.14
12192	-3.38	-2.85	-0.53
12784	-3.32	-2.61	-0.71
13818	-3.20	-2.46	-0.74
14970	-2.76	-2.39	-0.37

TABLE 4.3-9

V.C. SUMMER NUCLEAR STATION CYCLE 5
AXIAL OFFSET COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

Cycle Burnup MWD/MTU	Axial Offset (%) M	P	(M-P)
423	3.82	3.15	0.67
2201	2.62	0.41	2.21
3260	0.25	-1.13	1.38
4090	-1.67	-1.98	0.31
5064	-1.58	-2.80	1.22
6174	-2.38	-3.18	0.80
7438	-2.68	-3.33	0.65
8168	-1.78	-3.38	1.60
9354	-2.29	-3.46	1.17
10539	-4.05	-3.38	-0.67
11647	-2.68	-3.29	0.61
12793	-2.89	-3.30	0.41

"This page intentionally blank."

FIGURE 4.1-1

V.C. SUMMER NUCLEAR STATION CYCLE 3
CORE LOADING

	H	G	F	E	D	C	B	A
8	1	5 16	4 C	3	4	4A 40(a)	4 D	2
9	5 16	3 SB	5 16	3	5 24	4 SA	5 12	4
10	4 C	5 16	3 D	5 20	4 B	5 24	5 A	
11	3	3	5 20	3 SB	5 16	5 4	3	
12	4	5 24	4 B	5 16	4 C	3		
13	4A 40(a)	4 SA	5 24	5 4	3			
14	4 D	5 12	5 A	3				
15	2	4						

LEGEND

BATCH
# WABA's
RCCA(b)

(a) Depleted IFBA Rods

(b) Control Rod Banks (D,C,B,A) or Shutdown Banks (SB,SA)

FIGURE 4.1-2

V.C. SUMMER NUCLEAR STATION CYCLE 4
CORE LOADING

	H	G	F	E	D	C	B	A
8	4	5 16	5 C	5	4	5 20	5 D	4
9	5 16	4 SB	6 20	5	6 20	5 SA	6 4	5
10	5 C	6 20	4 D	6 16	5 B	6 12	5 A	
11	5	5	6 16	4A 40(a) SB	6 8	6 8	5	
12	4	6 20	5 B	6 8	4 C	5		
13	5 20	5 SA	6 12	6 8	5			
14	5 D	6 4	5 A	5				
15	4	5						

LEGEND

BATCH
# WABA's 	
RCCA(b)

(a) Depleted IFBA Rods

(b) Control Rod Banks (D,C,B,A) or Shutdown Banks (SB,SA)

FIGURE 4.1-3

V.C. SUMMER NUCLEAR STATION CYCLE 5
CORE LOADING

	H	G	F	E	D	C	B	A
8	3	7A 64	5 C	5	5	7B 64	4 D	6
9	7A 64	5 SB	7A 64	6	6	6 SA	7B 64	6
10	5 C	7A 64	6 D	7A 80	6 B	7A 80	7B A	
11	5	6	7A 80	5 SB	7A 80	7B	5	
12	5	6	6 B	7A 80	5 C	5		
13	7B 64	6 SA	7A 80	7B	5			
14	4 D	7B 64	7B A	5				
15	6	6						

LEGEND

BATCH
IFBA's
RCCA[(a)]

(a) Control Rod Banks (D,C,B,A) or Shutdown Rod Banks (SB,SA)

FIGURE 4.2-1

V.C. SUMMER NUCLEAR STATION CYCLE 3
MEASURED VERSUS PREDICTED
BANK B INTEGRAL ROD WORTH

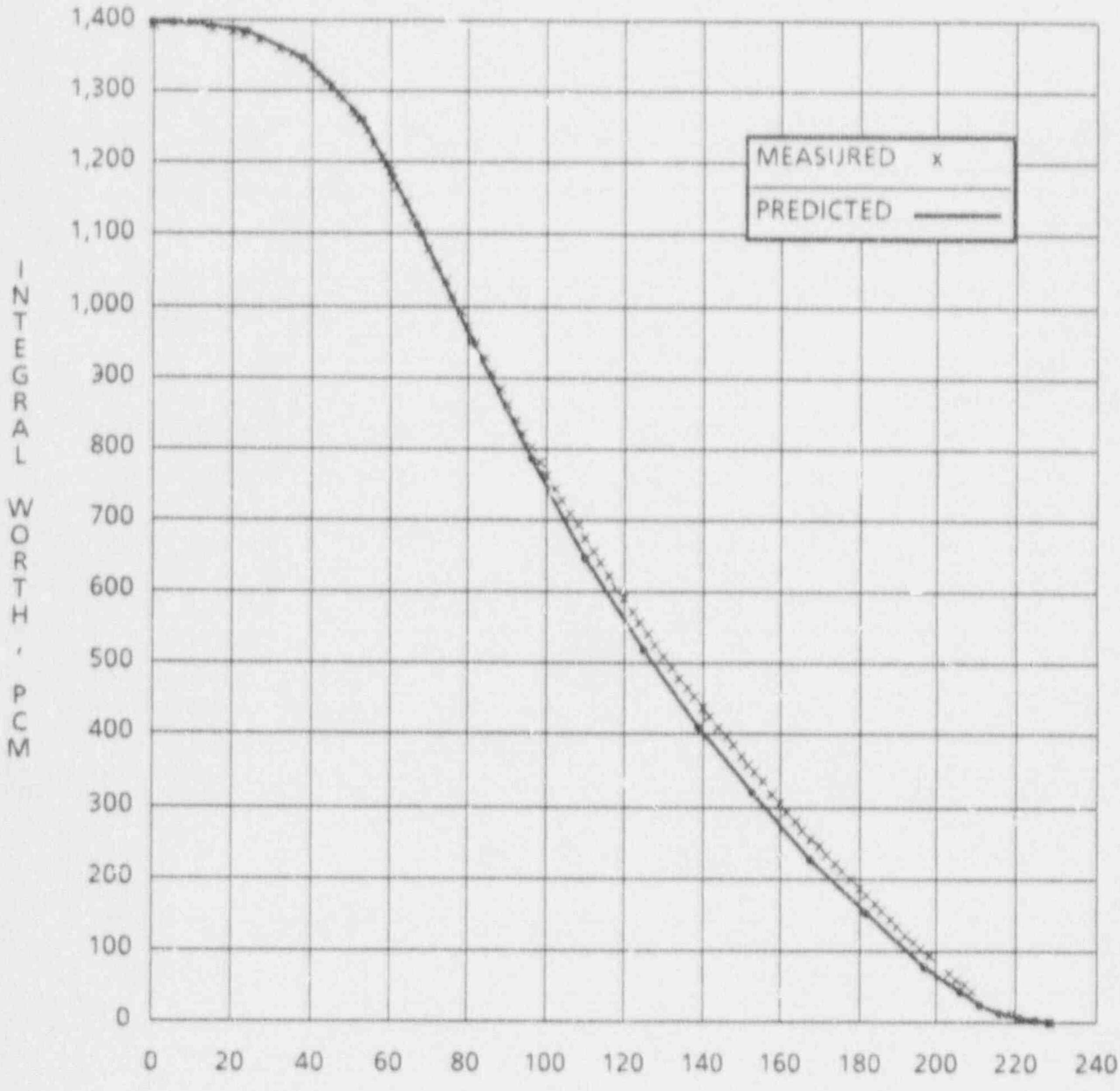


FIGURE 4.2-2

V.C. SUMMER NUCLEAR STATION CYCLE 4
MEASURED VERSUS PREDICTED
BANK B INTEGRAL ROD WORTH

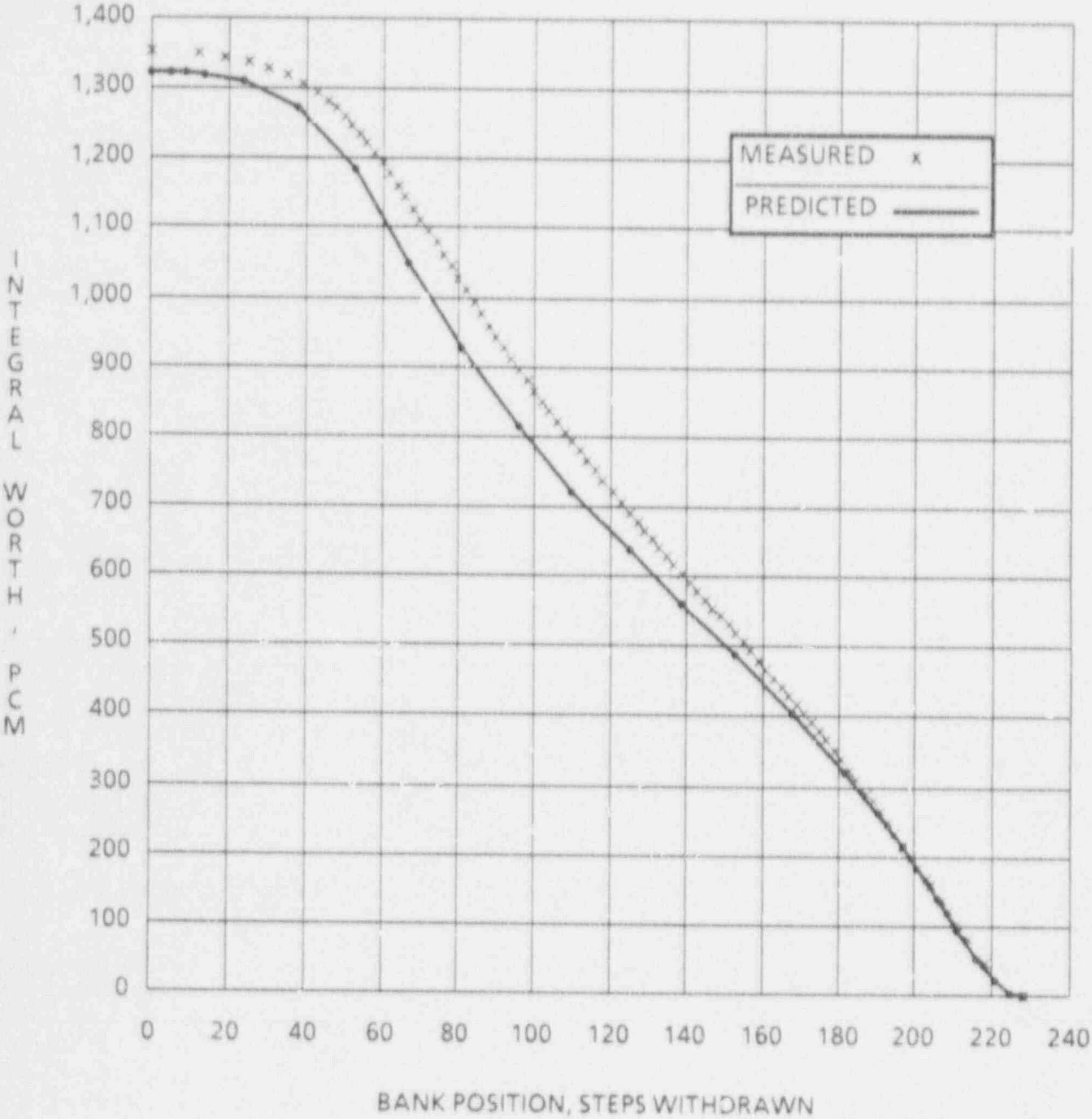


FIGURE 4.2-3

V.C. SUMMER NUCLEAR STATION CYCLE 5
MEASURED VERSUS PREDICTED
BANK B INTEGRAL ROD WORTH

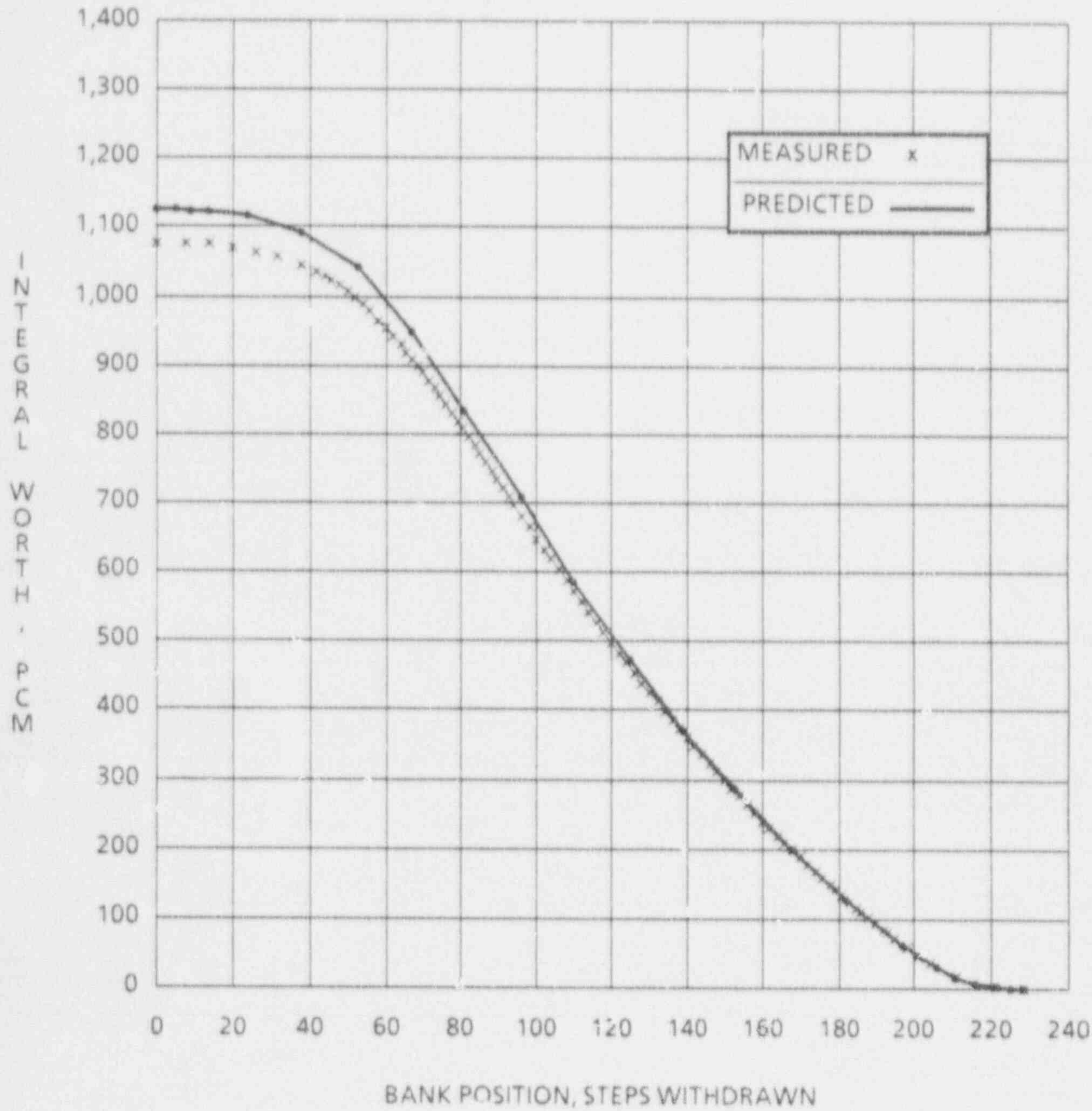


FIGURE 4.3-1

V.C. SUMMER NUCLEAR STATION CYCLE 3
BORON LETDOWN COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

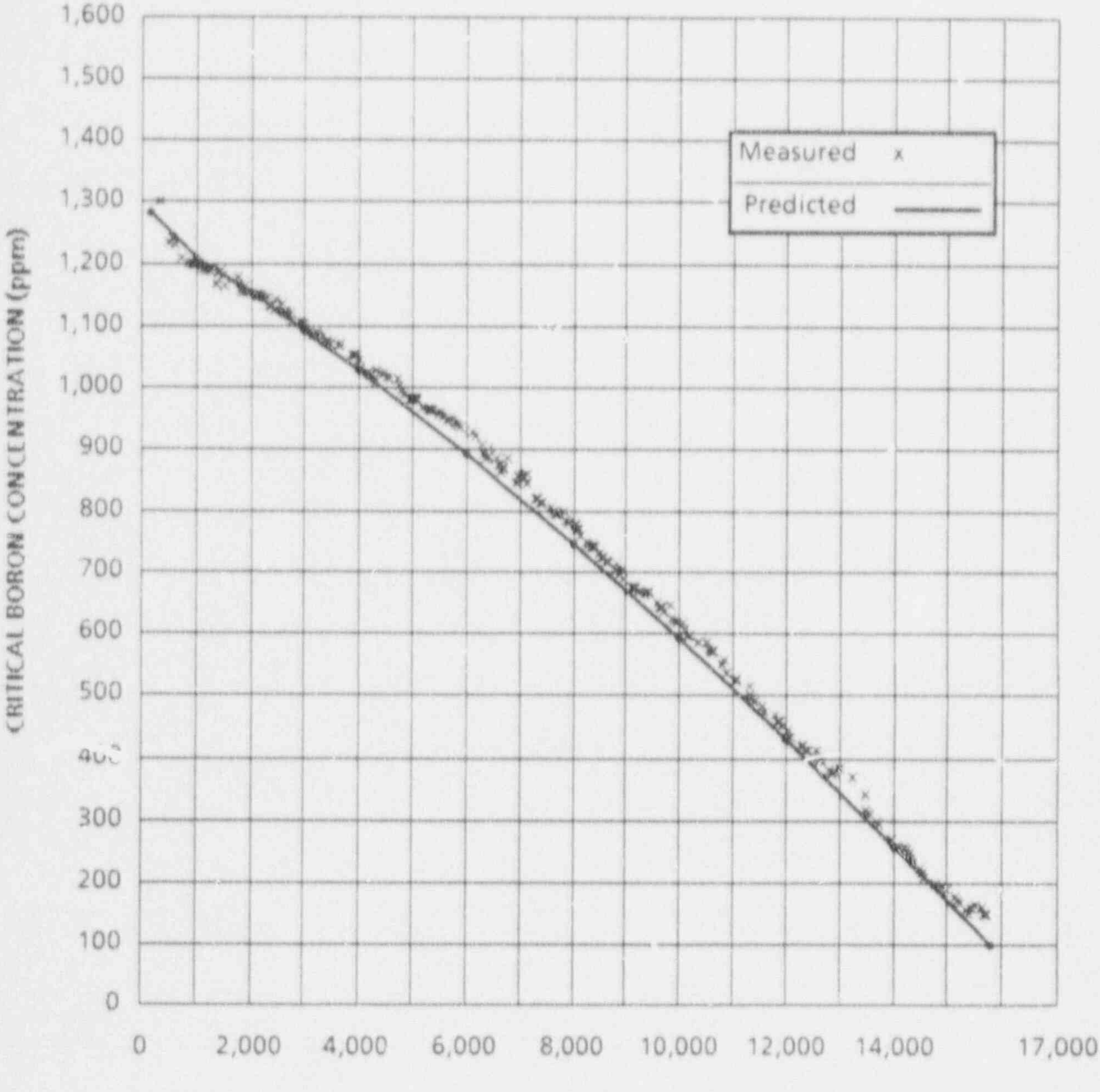


FIGURE 4.3-2

V.C. SUMMER NUCLEAR STATION CYCLE 4
BORON LETDOWN COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

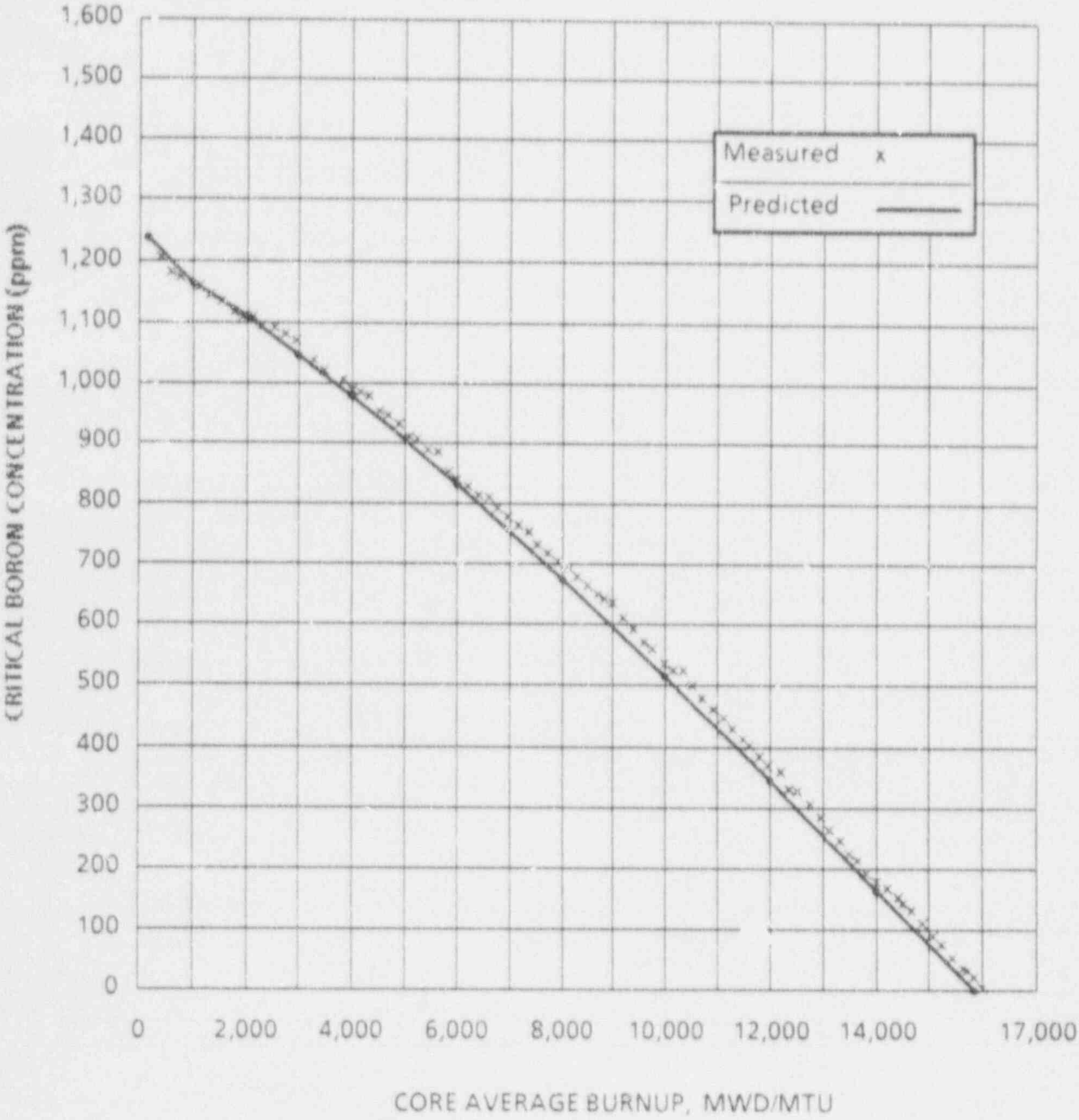


FIGURE 4.3-3

V.C. SUMMER NUCLEAR STATION CYCLE 5
BORON LETDOWN COMPARISON
BETWEEN MEASUREMENT AND PREDICTION

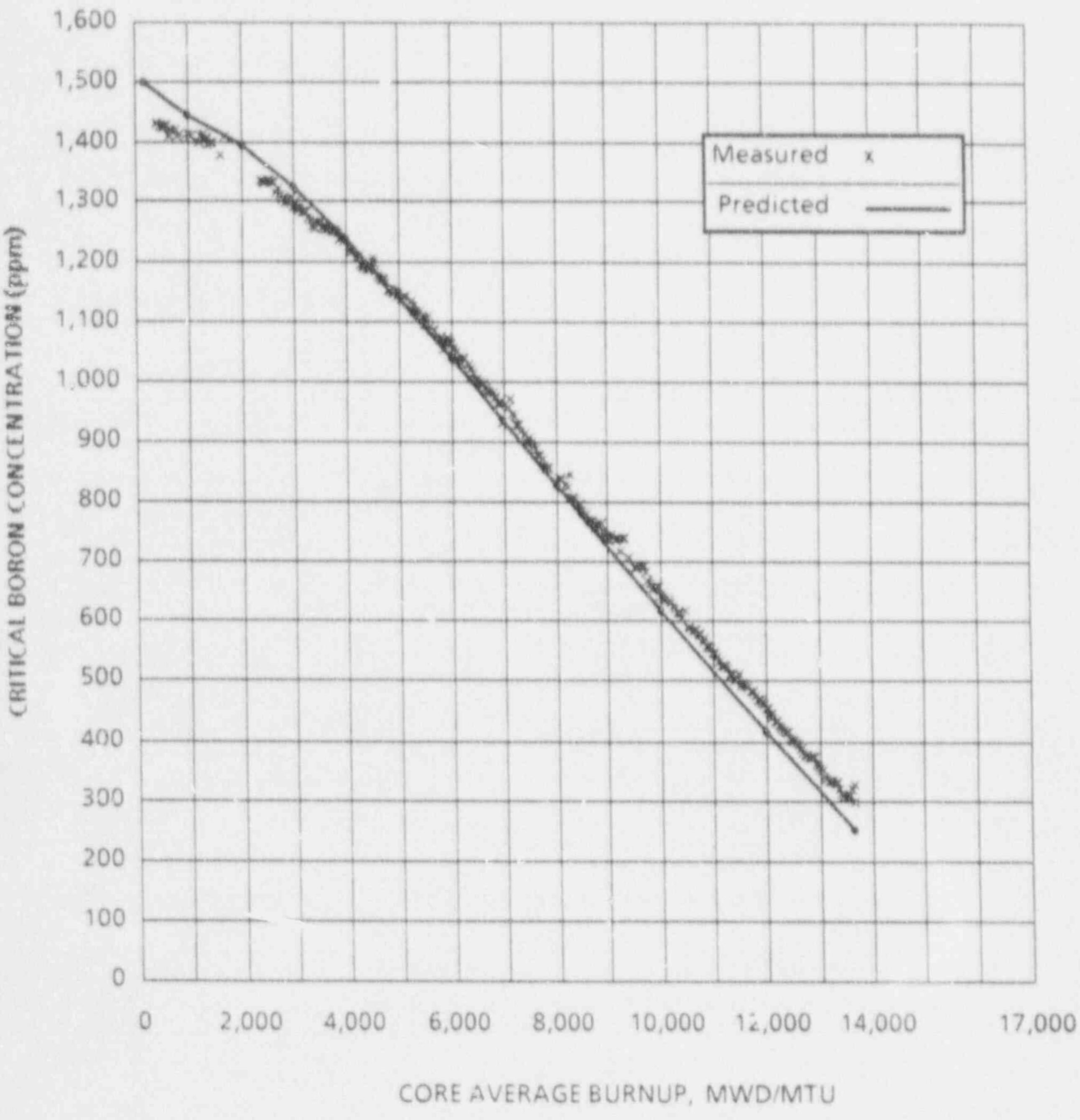


FIGURE 4.3-4

V.C. SUMMER NUCLEAR STATION CYCLE 3
F-DELTA-H COMPARISON
BETWEEN INCORE AND ANC

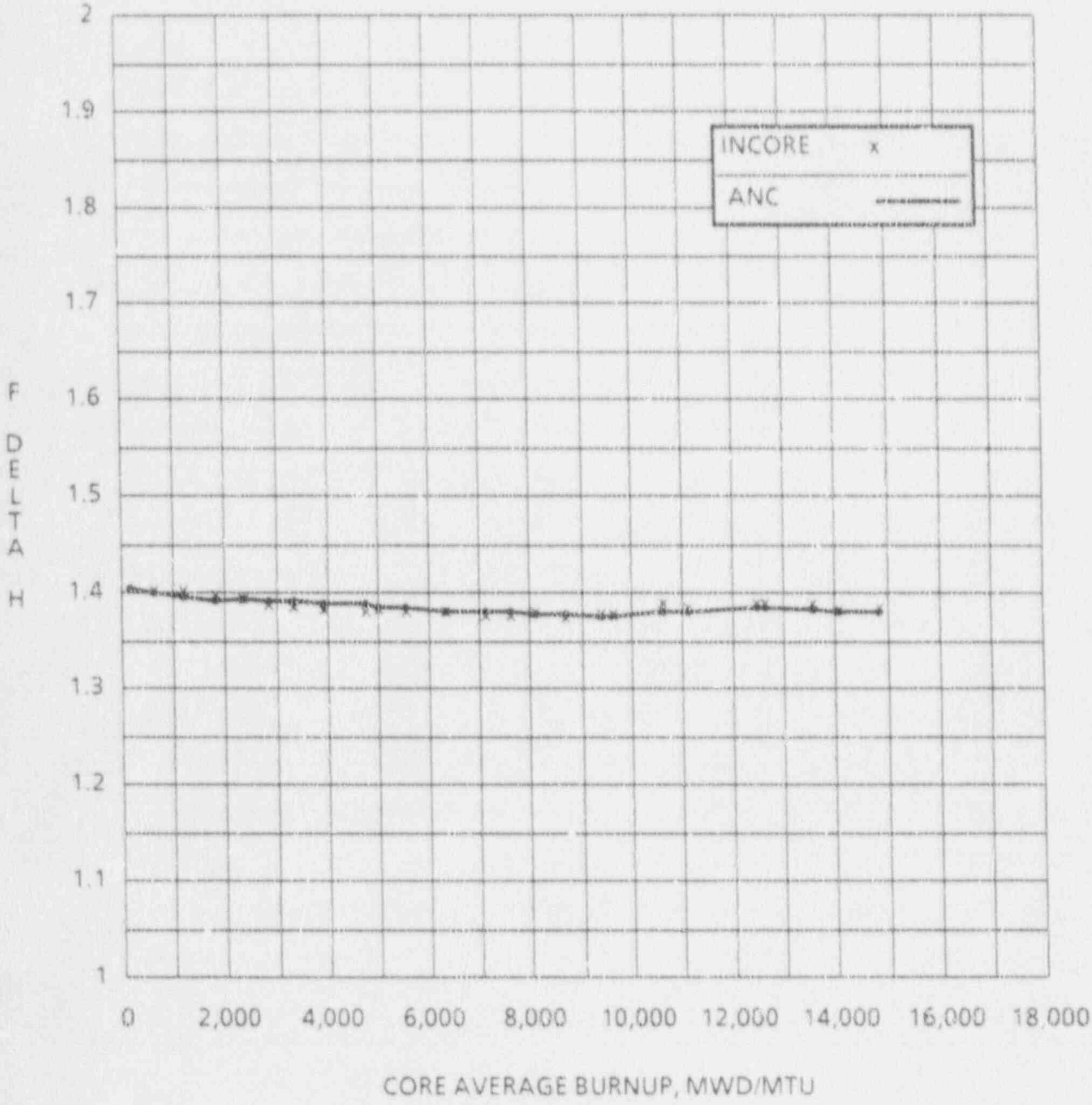


FIGURE 4.3-5

V.C. SUMMER NUCLEAR STATION CYCLE 4
F-DELTA-H COMPARISON
BETWEEN INCORE AND ANC

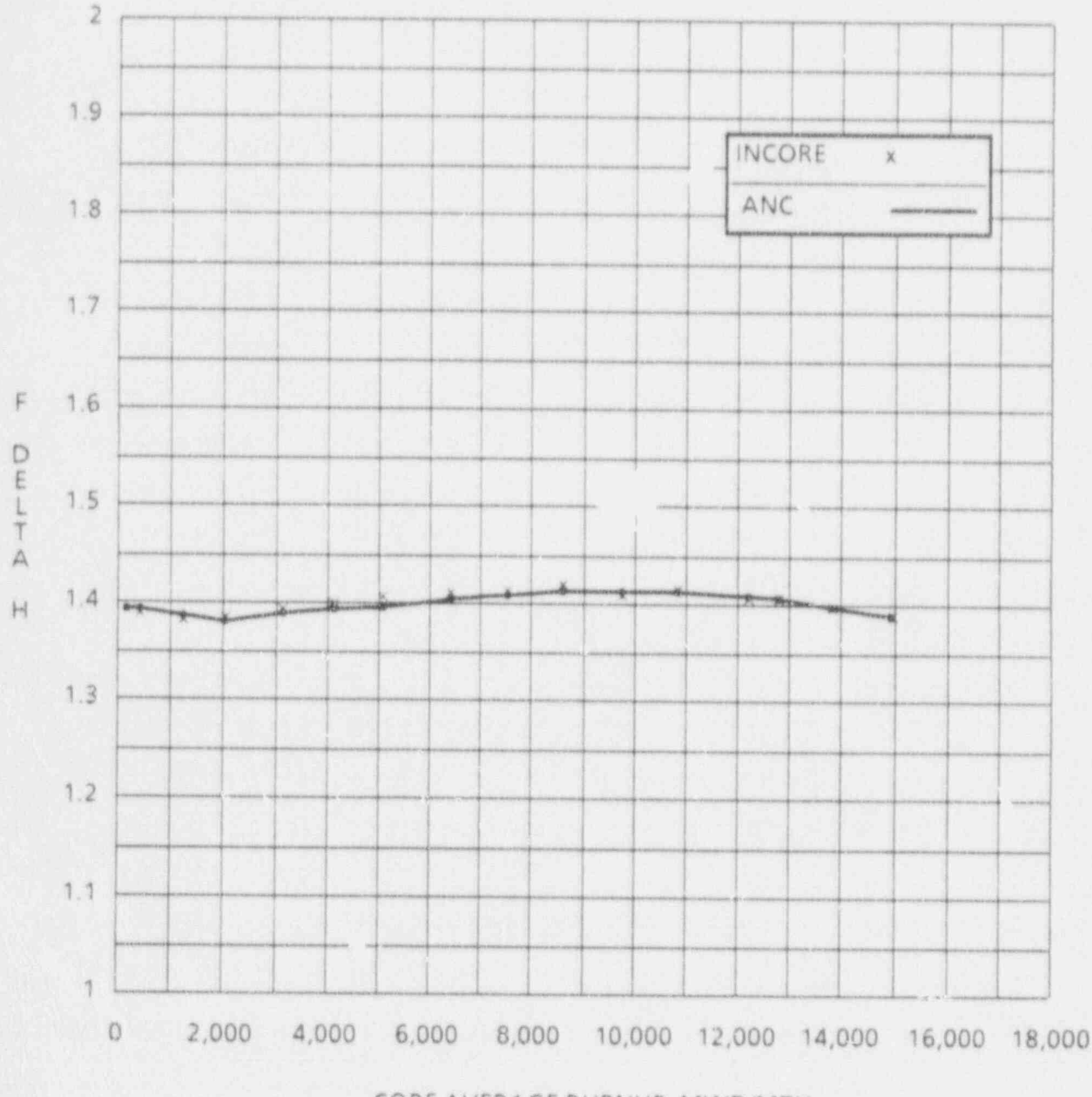
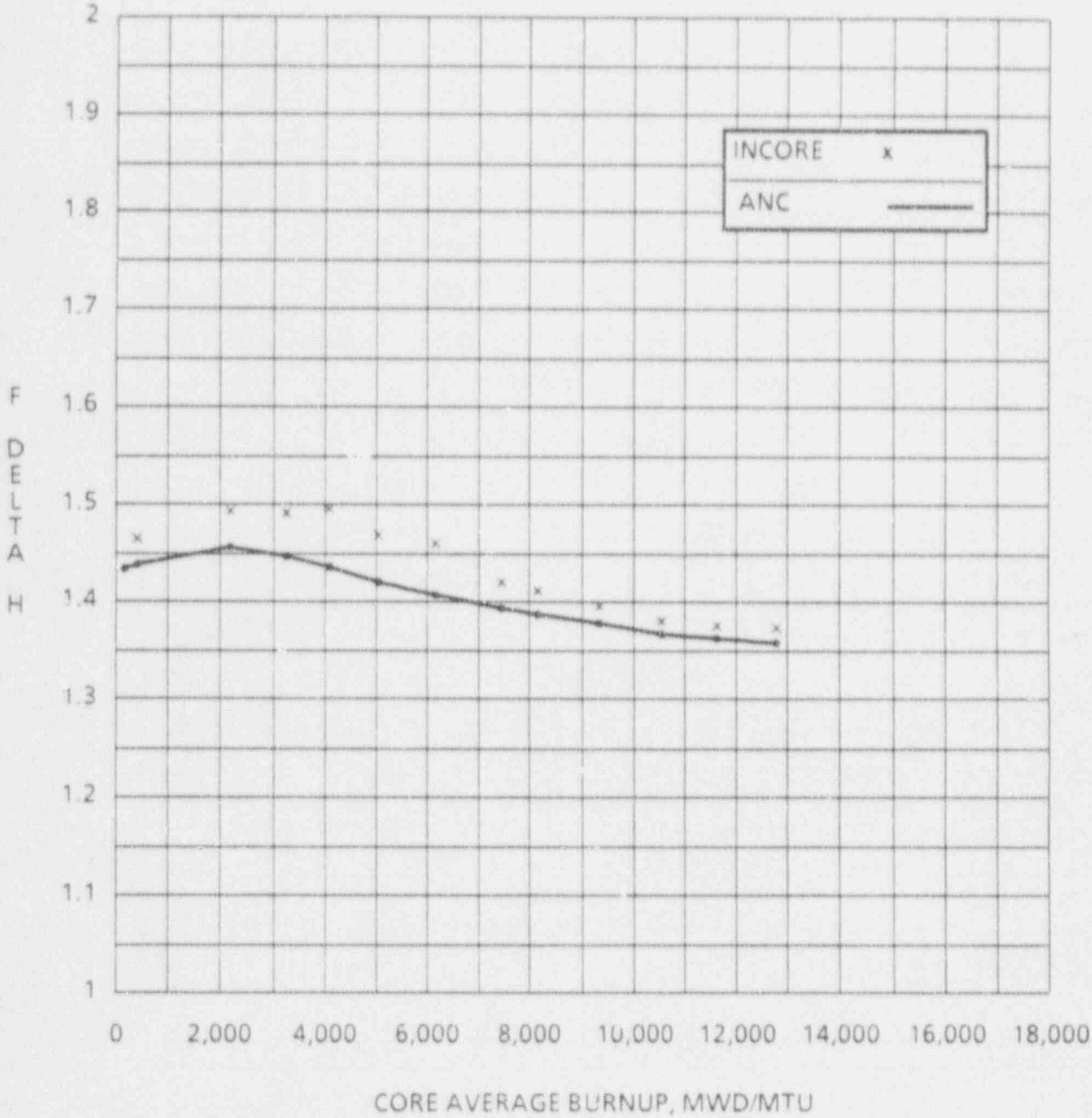


FIGURE 4.3-6

V.C. SUMMER NUCLEAR STATION CYCLE 5
F-DELTA-H COMPARISON
BETWEEN INCORE AND ANC



F'GURE 4.3-7

V.C. SUMMER NUCLEAR STATION CYCLE 3
Fq COMPARISON
BETWEEN INCORE AND ANC

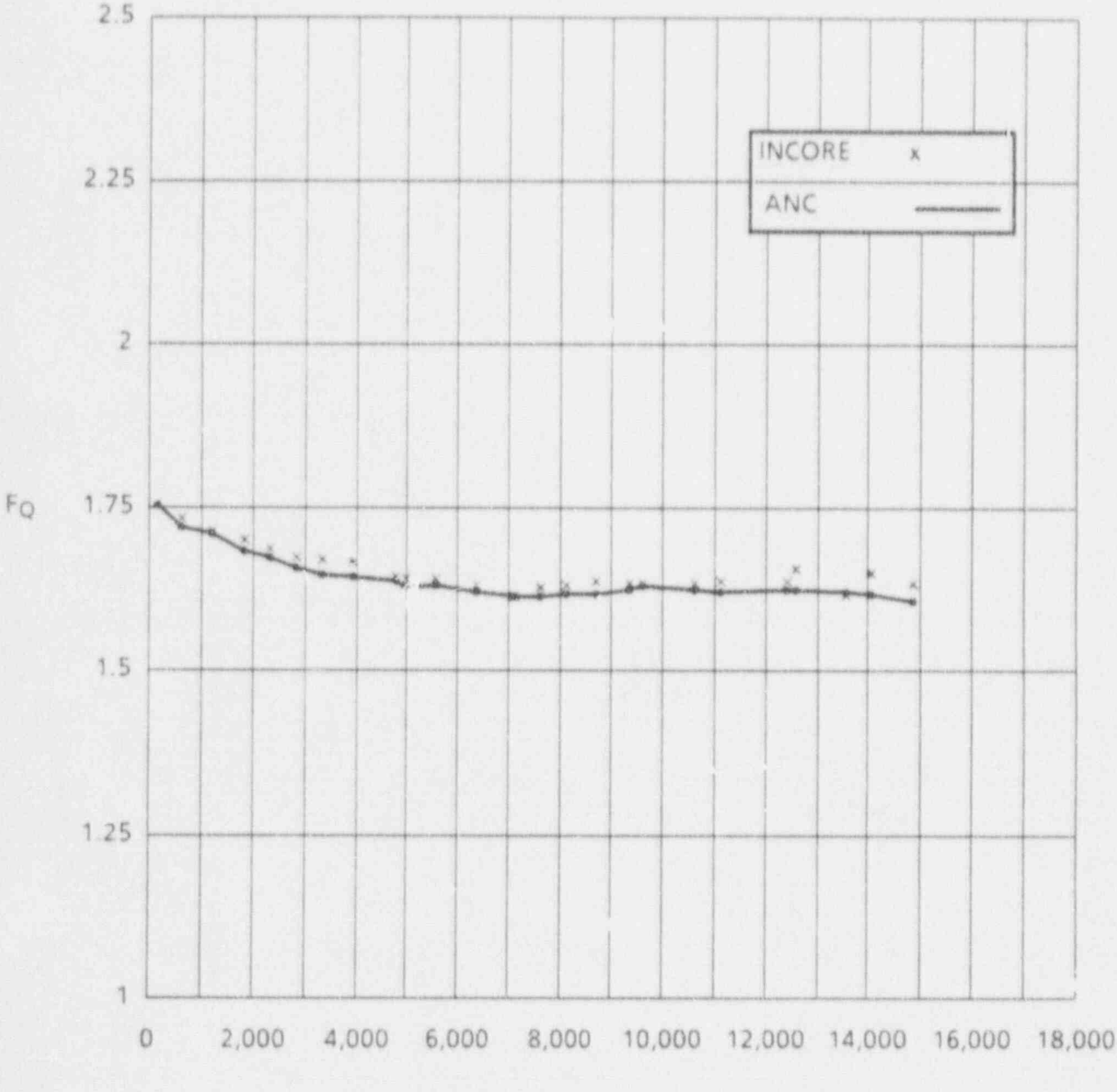


FIGURE 4.3-8

V.C. SUMMER NUCLEAR STATION CYCLE 4
Fq COMPARISON
BETWEEN INCORE AND ANC

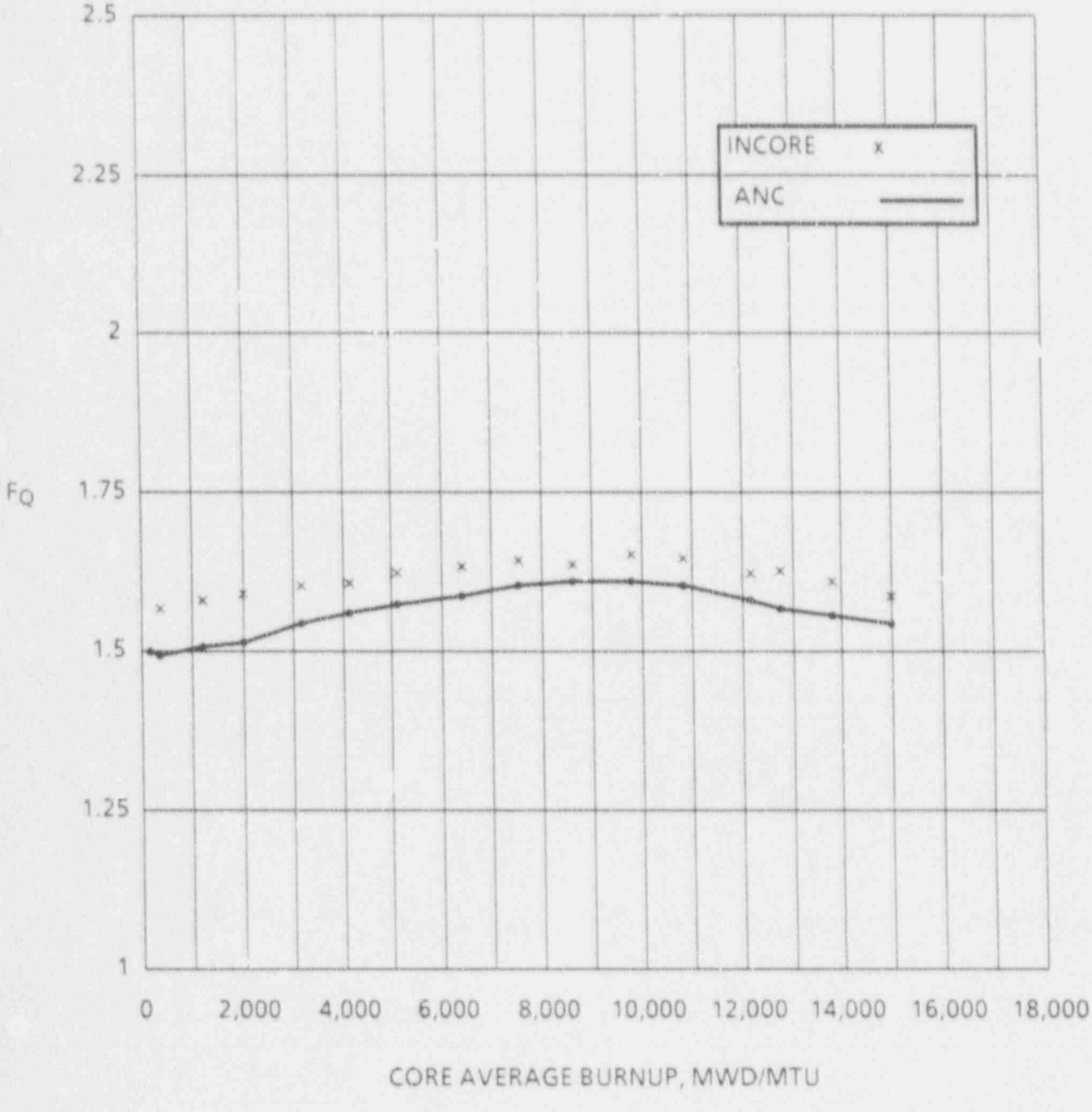


FIGURE 4.3-9

V.C. SUMMER NUCLEAR STATION CYCLE 5
Fq COMPARISON
BETWEEN INCORE AND ANC

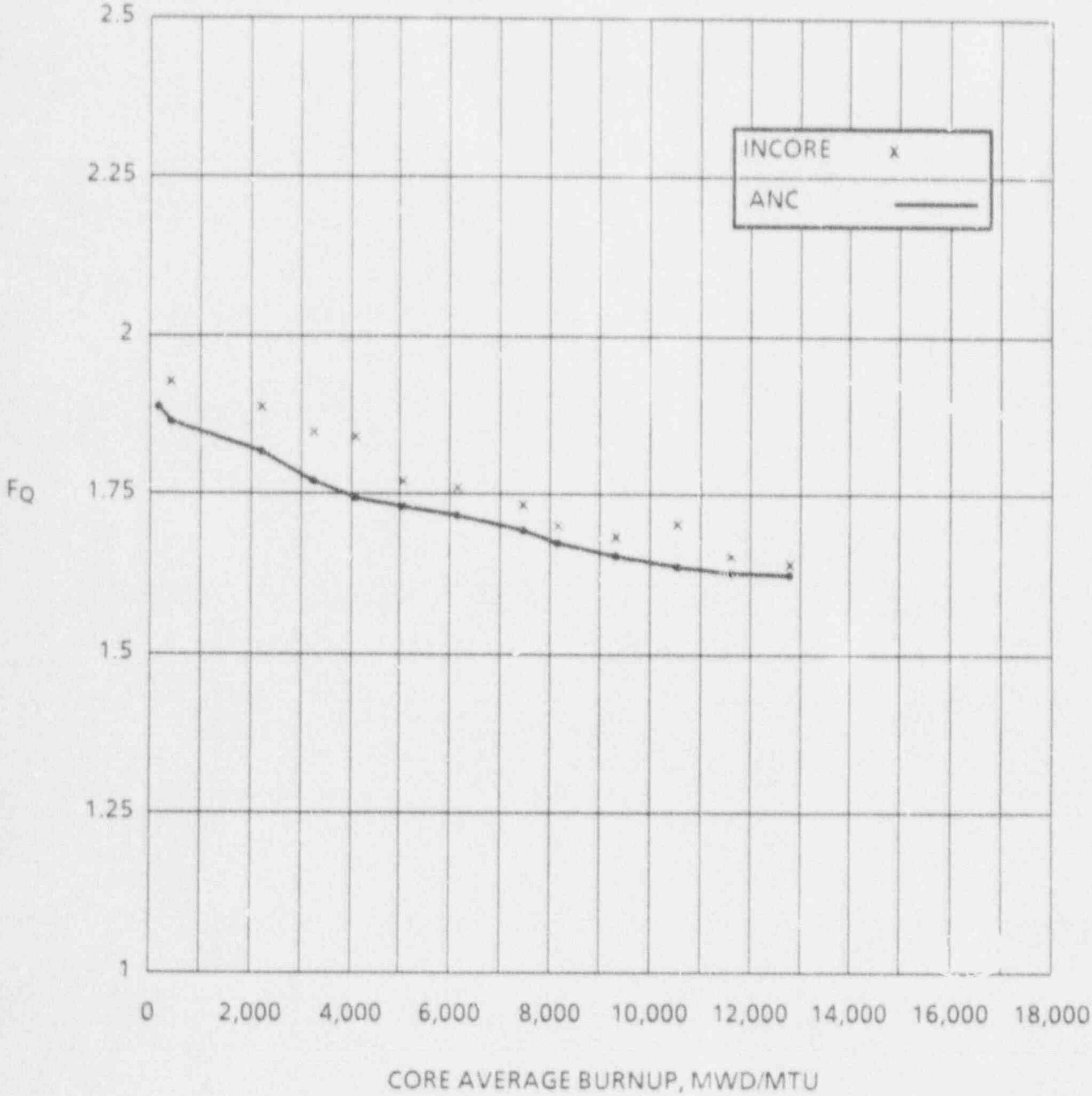


FIGURE 4.3-10

V.C. SUMMER NUCLEAR STATION CYCLE 3
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-03-007

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.427 0.419 1.91	0.402 0.395 1.77	0.435 0.422 3.08						
2					0.415 0.412 0.73	0.924 0.926 -0.22	1.040 1.032 0.78	1.027 1.019 0.79	1.061 1.038 2.22	0.967 0.929 4.09	0.425 0.413 2.91				
3				0.424 0.420 0.95	1.066 1.055 1.04	1.140 1.135 0.44	1.239 1.238 0.08	1.206 1.203 0.25	1.257 1.242 1.21	1.164 1.138 2.28	1.079 1.057 2.08	0.427 0.420 1.67			
4			0.428 0.420 1.90	0.874 0.864 1.16	1.188 1.180 0.68	1.261 1.266 -0.39	1.207 1.215 -0.66	1.257 1.264 -0.55	1.223 1.219 0.33	1.270 1.269 0.08	1.191 1.182 0.76	0.871 0.864 0.81	0.426 0.420 1.43		
5		0.425 0.413 2.91	1.088 1.057 2.93	1.197 1.182 1.27	1.021 1.015 0.59	1.188 1.190 -0.17	1.017 1.037 -1.93	1.056 1.077 -1.95	1.030 1.045 -1.44	1.184 1.196 -1.00	1.011 1.015 -0.39	1.186 1.180 0.51	1.070 1.055 1.42	0.418 0.412 1.46	
6		0.938 0.929 0.97	1.149 1.138 0.97	1.277 1.269 0.63	1.201 1.196 0.42	0.983 0.994 -1.11	1.205 1.239 -2.74	1.227 1.264 -2.93	1.207 1.247 -3.21	0.969 0.994 -2.52	1.188 1.190 -0.17	1.278 1.266 0.95	1.150 1.135 1.32	0.937 0.926 1.19	
7	0.422 0.422 0.00	1.034 1.038 -0.39	1.235 1.242 -0.56	1.213 1.219 -0.49	1.040 1.045 -0.48	1.232 1.247 -1.20	1.025 1.054 -2.75	1.237 1.276 -3.06	1.022 1.054 -3.04	1.200 1.239 -3.15	1.025 1.037 -1.16	1.222 1.215 0.58	1.236 1.238 -0.16	1.025 1.032 -0.68	0.417 0.419 -0.48
8	0.395 0.395 0.00	1.015 1.019 -0.39	1.199 1.203 -0.33	1.247 1.264 -1.34	1.054 1.077 -2.14	1.235 1.264 -2.29	1.241 1.276 -2.74	0.949 0.976 -2.77	1.252 1.276 -1.88	1.241 1.264 -1.82	1.07[illegible] 1.07[illegible] -0.09	1.269 1.264 [illegible]40	1.200 1.203 -0.25	1.010 1.019 -0.88	0.392 0.395 -0.76
9	0.419 0.419 0.00	1.038 1.032 0.58	1.249 1.238 0.89	1.207 1.215 -0.66	1.015 1.037 -2.12	1.221 1.239 -1.45	1.055 1.054 0.09	1.265 1.276 -0.86	1.042 1.054 -1.14	1.234 1.247 -1.04	1.038 1.045 -0.67	1.212 1.219 -0.57	1.239 1.242 -0.24	1.034 1.038 -0.39	0.422 0.422 0.00
10		0.943 0.926 1.84	1.156 1.135 1.85	1.263 1.266 -0.24	1.174 1.190 -1.34	0.986 0.994 -0.80	1.236 1.247 -0.88	1.249 1.264 -1.19	1.234 1.239 -0.40	0.998 0.994 0.40	1.204 1.196 0.67	1.271 1.269 0.16	1.153 1.138 1.32	0.941 0.929 1.29	
11		0.416 0.412 0.97	1.066 1.055 1.04	1.187 1.180 0.59	1.014 1.015 -0.10	1.198 1.196 0.17	1.040 1.045 -0.48	1.071 1.077 -0.56	1.036 1.037 -0.10	1.203 1.190 1.09	1.022 1.015 0.69	1.187 1.182 0.42	1.070 1.057 1.23	0.424 0.413 2.66	
12			0.420 0.420 0.00	0.864 0.864 0.00	1.180 1.182 -0.17	1.272 1.269 0.24	1.212 1.219 -0.57	1.256 1.264 -0.63	1.221 1.215 0.49	1.288 1.266 1.74	1.197 1.180 1.44	0.869 0.864 0.58	0.423 0.420 0.71		
13				0.420 0.420 0.00	1.056 1.057 -0.09	1.140 1.138 0.18	1.242 1.242 0.00	1.210 1.203 0.58	1.256 1.238 1.45	1.163 1.135 2.47	1.075 1.055 1.90	0.423 0.420 0.71			
14					0.412 0.413 -0.24	0.936 0.929 0.75	1.044 1.038 0.58	1.029 1.019 0.98	1.047 1.032 1.45	0.949 0.926 2.48	0.424 0.412 2.91				
15							0.429 0.422 1.66	0.402 0.395 1.77	0.427 0.419 1.91						

Mean Absolute Difference = 0.0107
Standard Deviation = 0.0142

INCORE ANC % DIFFERENCE

BURNUP = 1839 MWD/MTU   POWER LEVEL = 100.0%   D BANK AT 228 STEPS

FIGURE 4.3-11

V.C. SUMMER NUCLEAR STATION CYCLE 3
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-03-014

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.420 0.413 1.69	0.394 0.387 1.81	0.426 0.415 2.65						
2					0.418 0.415 0.72	0.911 0.917 -0.65	1.024 1.017 0.69	0.982 0.974 0.82	1.043 1.022 2.05	0.950 0.919 3.37	0.430 0.416 3.37				
3				0.429 0.426 0.70	1.070 1.063 0.66	1.160 1.168 -0.68	1.199 1.198 0.08	1.143 1.139 0.35	1.218 1.201 1.42	1.209 1.170 3.33	1.100 1.065 3.29	0.432 0.427 1.17			
4			0.431 0.427 0.94	0.876 0.870 0.69	1.215 1.211 0.33	1.246 1.258 -0.95	1.235 1.241 -0.48	1.215 1.220 -0.41	1.251 1.245 0.48	1.258 1.260 -0.16	1.221 1.214 0.58	0.875 0.870 0.57	0.433 0.426 1.64		
5		0.422 0.416 1.44	1.080 1.065 1.41	1.223 1.214 0.74	1.029 1.025 0.39	1.228 1.233 -0.41	1.014 1.030 -1.55	1.036 1.052 -1.52	1.018 1.037 -1.83	1.221 1.239 -1.45	1.020 1.025 -0.49	1.219 1.211 0.66	1.080 1.063 1.60	0.422 0.415 1.69	
6		0.922 0.919 0.33	1.174 1.170 0.34	1.265 1.260 0.40	1.245 1.239 0.48	0.995 1.002 -0.70	1.235 1.261 -2.06	1.213 1.238 -2.02	1.239 1.269 -2.36	0.985 1.002 -1.70	1.226 1.233 -0.57	1.266 1.258 0.64	1.185 1.168 1.46	0.931 0.917 1.53	
7	0.412 0.415 -0.72	1.018 1.022 -0.39	1.196 1.201 -0.42	1.241 1.245 -0.32	1.034 1.037 -0.29	1.259 1.269 -0.79	1.029 1.050 -2.00	1.272 1.298 -2.00	1.030 1.050 -1.90	1.238 1.261 -1.82	1.012 1.030 -1.75	1.230 1.241 -0.89	1.193 1.198 -0.42	1.014 1.017 -0.29	0.412 0.413 -0.24
8	0.388 0.387 0.26	0.971 0.974 -0.31	1.138 1.139 -0.09	1.207 1.220 -1.07	1.034 1.052 -1.71	1.217 1.238 -1.70	1.276 1.298 -1.69	0.962 0.976 -1.43	1.290 1.298 -0.62	1.231 1.238 -0.57	1.053 1.052 0.10	1.210 1.220 -0.82	1.135 1.139 -0.35	0.971 0.974 -0.31	0.387 0.387 0.00
9	0.422 0.413 2.18	1.028 1.017 1.03	1.211 1.198 1.09	1.237 1.241 -0.32	1.012 1.030 -1.75	1.246 1.261 -1.19	1.052 1.050 0.19	1.294 1.298 -0.31	1.048 1.050 -0.19	1.267 1.269 -0.16	1.035 1.037 -0.19	1.236 1.245 -0.72	1.188 1.201 -1.08	1.014 1.022 -0.78	0.413 0.415 -0.48
10		0.936 0.917 2.07	1.191 1.168 1.97	1.257 1.258 -0.08	1.218 1.233 1.22	0.995 1.002 -0.70	1.263 1.269 -0.47	1.231 1.238 -0.57	1.256 1.261 -0.40	1.004 1.002 0.20	1.241 1.239 0.16	1.258 1.260 -0.16	1.164 1.170 -0.51	0.921 0.919 0.22	
11		0.419 0.415 0.96	1.074 1.063 1.03	1.217 1.211 0.50	1.020 1.025 -0.49	1.240 1.239 0.08	1.035 1.037 -0.19	1.050 1.052 -0.19	1.018 1.030 -1.17	1.237 1.233 0.32	1.027 1.025 0.20	1.217 1.214 0.25	1.067 1.065 0.19	0.417 0.416 0.24	
12			0.426 0.426 0.00	0.867 0.870 -0.34	1.208 1.214 -0.49	1.261 1.260 0.08	1.242 1.245 -0.24	1.217 1.220 -0.25	1.250 1.241 0.73	1.277 1.258 1.51	1.227 1.211 1.32	0.876 0.870 0.69	0.431 0.427 0.94		
13				0.426 0.427 -0.23	1.061 1.065 -0.38	1.171 1.170 0.09	1.207 1.201 0.50	1.147 1.139 0.70	1.219 1.198 1.75	1.189 1.168 1.80	1.083 1.063 1.88	0.430 0.426 0.94			
14					0.414 0.416 -0.48	0.916 0.919 -0.33	1.019 1.022 -0.29	0.983 0.974 0.92	1.034 1.017 1.67	0.933 0.917 1.74	0.426 0.415 2.65				
15							0.414 0.415 -0.24	0.391 0.387 1.03	0.416 0.413 0.73						

Mean Absolute Difference = 0.0088
Standard Deviation = 0.0117

INCORE ANC % DIFFERENCE

BURNUP = 3920 MWD/MTU   POWER LEVEL = 99.0%   D BANK AT 228 STEPS

FIGURE 4.3-12

V.C. SUMMER NUCLEAR STATION CYCLE 3
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-03-017

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.420 0.411 2.19	0.394 0.386 2.07	0.423 0.414 2.17						
2					0.424 0.418 1.44	0.905 0.910 -0.55	1.018 1.010 0.79	0.960 0.952 0.84	1.033 1.015 1.77	0.932 0.912 2.19	0.428 0.418 2.39				
3				0.437 0.431 1.39	1.080 1.065 1.41	1.181 1.187 -0.51	1.174 1.175 -0.09	1.105 1.104 0.09	1.188 1.178 0.85	1.216 1.190 2.18	1.090 1.066 2.25	0.437 0.431 1.39			
4			0.436 0.431 1.16	0.877 0.873 0.46	1.232 1.227 0.41	1.242 1.252 -0.80	1.254 1.258 -0.32	1.192 1.196 -0.33	1.263 1.261 0.16	1.245 1.253 -0.64	1.232 1.229 0.24	0.876 0.873 0.34	0.437 0.431 1.39		
5		0.421 0.418 0.72	1.073 1.066 0.66	1.233 1.229 0.33	1.031 1.030 0.10	1.256 1.260 -0.32	1.015 1.027 -1.17	1.027 1.040 -1.25	1.020 1.033 -1.26	1.254 1.265 -0.87	1.028 1.030 -0.19	1.232 1.227 0.41	1.080 1.065 1.41	0.424 0.418 1.44	
6		0.910 0.912 -0.22	1.187 1.190 -0.25	1.253 1.253 0.00	1.267 1.265 0.16	1.003 1.007 -0.40	1.254 1.271 -1.34	1.202 1.220 -1.48	1.260 1.279 -1.49	0.997 1.007 -0.99	1.257 1.260 -0.24	1.256 1.252 0.32	1.200 1.187 1.10	0.919 0.910 0.99	
7	0.409 0.414 -1.21	1.009 1.015 -0.59	1.171 1.178 -0.59	1.257 1.261 -0.32	1.032 1.033 -0.10	1.274 1.279 -0.39	1.030 1.046 -1.53	1.287 1.306 -1.45	1.031 1.046 -1.43	1.256 1.271 -1.18	1.015 1.027 -1.17	1.258 1.258 0.00	1.169 1.175 -0.51	1.005 1.010 -0.50	0.408 0.411 -0.73
8	0.386 0.386 0.00	0.947 0.952 -0.53	1.100 1.104 -0.36	1.184 1.196 -1.00	1.024 1.040 -1.54	1.203 1.220 -1.39	1.292 1.306 -1.07	0.964 0.974 -1.03	1.300 1.306 -0.46	1.214 1.220 -0.49	1.040 1.040 0.00	1.196 1.196 0.00	1.099 1.104 -0.45	0.948 0.952 -0.42	0.384 0.386 -0.52
9	0.417 0.411 1.46	1.018 1.010 0.79	1.184 1.175 0.77	1.253 1.258 -0.40	1.011 1.027 -1.56	1.259 1.271 -0.94	1.047 1.046 0.10	1.303 1.306 -0.23	1.044 1.046 -0.19	1.278 1.279 -0.08	1.035 1.033 0.19	1.263 1.261 0.16	1.171 1.178 -0.59	1.009 1.015 -0.59	0.411 0.414 -0.72
10		0.922 0.910 1.32	1.204 1.187 1.43	1.250 1.252 -0.16	1.249 1.260 -0.87	1.001 1.007 -0.60	1.277 1.279 -0.16	1.215 1.220 -0.41	1.273 1.271 0.16	1.010 1.007 0.30	1.272 1.265 0.55	1.255 1.253 0.16	1.193 1.190 0.25	0.917 0.912 0.55	
11		0.421 0.418 0.72	1.073 1.065 0.75	1.235 1.227 0.65	1.032 1.030 0.19	1.264 1.265 -0.08	1.030 1.033 -0.29	1.036 1.040 -0.38	1.022 1.027 -0.49	1.260 1.260 0.00	1.032 1.030 0.19	1.231 1.229 0.16	1.069 1.066 0.28	0.421 0.418 0.72	
12			0.432 0.431 0.23	0.874 0.873 0.11	1.232 1.229 0.24	1.251 1.253 -0.16	1.255 1.261 -0.48	1.191 1.196 -0.42	1.268 1.258 0.79	1.265 1.252 1.04	1.237 1.227 0.81	0.874 0.873 -0.11	0.432 0.431 0.23		
13				0.432 0.431 0.23	1.069 1.066 0.28	1.188 1.190 -0.17	1.172 1.178 -0.51	1.101 1.104 -0.27	1.188 1.175 1.11	1.201 1.187 1.18	1.078 1.065 1.22	0.432 0.431 0.23			
14					0.420 0.418 0.48	0.914 0.912 0.22	1.018 1.015 0.30	0.952 0.952 0.00	1.020 1.010 0.99	0.919 0.910 0.99	0.427 0.418 2.15				
15							0.415 0.414 0.24	0.386 0.386 0.00	0.411 0.411 0.00						

Mean Absolute Difference = 0.0065
Standard Deviation = 0.0085

INCORE
ANC
% DIFFERENCE

BURNUP = 5558 MWD/MTU POWER LEVEL = 99.8% D BANK AT 228 STEPS

FIGURE 4.3-13

V.C. SUMMER NUCLEAR STATION CYCLE 3
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-03-025

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.409 0.415 -1.45	0.390 0.390 0.00	0.426 0.417 2.16						
2					0.428 0.424 0.94	0.889 0.899 -1.11	0.999 1.005 -0.60	0.919 0.925 -0.65	1.020 1.008 1.19	0.920 0.900 2.22	0.433 0.424 2.12				
3				0.442 0.440 0.45	1.068 1.064 0.38	1.210 1.219 -0.74	1.137 1.143 -0.52	1.051 1.054 -0.28	1.149 1.145 0.35	1.231 1.220 0.90	1.079 1.064 1.41	0.449 0.440 2.05			
4			0.444 0.440 0.91	0.878 0.876 0.23	1.248 1.248 0.00	1.225 1.236 -0.89	1.276 1.285 -0.70	1.153 1.162 -0.77	1.285 1.288 -0.23	1.230 1.237 -0.57	1.255 1.250 0.40	0.883 0.876 0.80	0.447 0.440 1.59		
5		0.427 0.424 0.71	1.072 1.064 0.75	1.252 1.250 0.16	1.035 1.036 -0.10	1.292 1.297 -0.39	1.010 1.021 -1.08	1.010 1.022 -1.17	1.061 1.026 -0.97	1.288 1.302 -1.08	1.030 1.036 -0.58	1.249 1.248 0.08	1.077 1.064 1.22	0.430 0.424 1.42	
6		0.907 0.900 0.78	1.229 1.220 0.74	1.241 1.237 0.32	1.304 1.302 0.15	1.008 1.013 -0.49	1.270 1.285 -1.17	1.173 1.187 -1.18	1.273 1.291 -1.39	0.999 1.013 -1.38	1.297 1.297 0.00	1.245 1.236 0.73	1.232 1.219 1.07	0.907 0.899 0.89	
7	0.426 0.417 2.16	1.021 1.008 1.29	1.148 1.145 0.26	1.292 1.288 0.31	1.029 1.026 0.29	1.290 1.291 -0.08	1.025 1.036 -1.06	1.299 1.313 -1.07	1.022 1.036 -1.35	1.268 1.285 -1.32	1.019 1.021 -0.20	1.296 1.285 0.86	1.144 1.143 0.09	1.000 1.005 -0.50	0.412 0.415 -0.72
8	0.399 0.390 2.31	0.936 0.925 1.19	1.060 1.054 0.57	1.156 1.162 -0.52	1.013 1.022 -0.88	1.177 1.187 -0.84	1.303 1.313 -0.76	0.9[illegible]8 0.968 -1.03	1.307 1.313 -0.46	1.181 1.187 -0.51	1.028 1.022 0.59	1.169 1.162 0.60	1.056 1.054 0.19	0.923 0.925 -0.22	0.390 0.390 0.00
9	0.424 0.415 2.17	1.018 1.005 1.29	1.153 1.143 0.87	1.285 1.285 0.00	1.012 1.021 -0.88	1.280 1.285 -0.39	1.042 1.036 0.58	1.312 1.313 -0.08	1.033 1.036 -0.29	1.289 1.291 -0.15	1.025 1.026 -0.10	1.282 1.288 -0.47	1.141 1.144 -0.35	1.009 1.008 0.10	0.418 0.417 0.24
10		0.913 0.899 1.56	1.237 1.219 1.48	1.234 1.236 -0.1[illegible]	1.283 1.297 -1.08	1.007 1.013 -0.59	1.290 1.291 -0.08	1.183 1.187 -0.34	1.286 1.285 0.08	1.013 1.013 0.00	1.304 1.302 0.15	1.228 1.237 -0.73	1.222 1.220 0.16	0.909 0.900 1.00	
11		0.426 0.424 0.47	1.071 1.064 0.66	1.249 1.248 0.08	1.024 1.03[illegible] -1.16	1.300 1.302 -0.15	1.025 1.026 -0.10	1.020 1.022 -0.20	1.022 1.021 0.10	1.299 1.297 0.15	1.034 1.036 -0.19	1.244 1.250 -0.48	1.066 1.064 0.19	0.427 0.424 0.71	
12			0.439 0.440 -0.23	0.871 0.876 -0.57	1.237 1.250 -1.04	1.231 1.237 -0.49	1.284 1.288 -0.31	1.157 1.162 -0.43	1.289 1.285 0.31	1.242 1.236 0.49	1.252 1.248 0.32	0.872 0.876 -0.46	0.441 0.440 0.23		
13				0.440 0.440 0.00	1.064 1.064 0.00	1.220 1.220 0.00	1.144 1.145 -0.09	1.052 1.054 -0.19	1.145 1.143 0.17	1.224 1.219 0.41	1.071 1.064 0.66	0.441 0.440 0.23			
14					0.424 0.424 0.00	0.911 0.900 1.22	1.020 1.008 1.19	0.929 0.925 0.43	1.005 1.005 0.00	0.902 0.899 0.33	0.429 0.424 1.18				
15							0.427 0.417 2.40	0.394 0.390 1.03	0.412 0.415 -0.72						

Mean Absolute Difference = 0.0060
Standard Deviation = 0.0076

INCORE
ANC
% DIFFERENCE

BURNUP = 8689 MWD/MTU POWER LEVEL = 100.0% D BANK AT 228 STEPS

FIGURE 4.3-14

V.C. SUMMER NUCLEAR STATION CYCLE 3
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-03-031

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.433 0.424 2.12	0.409 0.401 2.00	0.434 0.426 1.88						
2					0.439 0.431 1.86	0.888 0.898 -1.11	1.014 1.010 0.40	0.924 0.920 0.43	1.027 1.013 1.38	0.914 0.899 1.67	0.440 0.431 2.09				
3				0.454 0.449 1.11	1.074 1.064 0.94	1.230 1.236 -0.49	1.121 1.129 -0.71	1.027 1.031 -0.39	1.130 1.130 0.00	1.241 1.236 0.40	1.076 1.064 1.13	0.460 0.449 2.45			
4			0.453 0.449 0.89	0.884 0.881 0.34	1.261 1.256 0.40	1.214 1.223 -0.74	1.285 1.295 -0.77	1.133 1.143 -0.87	1.290 1.297 -0.54	1.212 1.223 -0.90	1.259 1.257 0.16	0.886 0.881 0.57	0.456 0.449 1.56		
5		0.432 0.431 0.23	1.066 1.064 0.19	1.260 1.257 0.24	1.038 1.036 0.19	1.308 1.310 -0.15	1.003 1.016 -1.28	1.000 1.012 -1.19	1.007 1.020 -1.27	1.299 1.314 -1.14	1.027 1.036 -0.87	1.253 1.256 -0.24	1.075 1.064 1.03	0.439 0.431 1.86	
6		0.893 0.899 -0.67	1.229 1.236 -0.57	1.221 1.223 -0.16	1.319 1.314 0.38	1.009 1.012 -0.30	1.272 1.284 -0.93	1.153 1.165 -1.03	1.274 1.289 -1.16	1.000 1.012 -1.19	1.308 1.310 -0.15	1.228 1.223 0.41	1.250 1.236 1.13	0.910 0.898 1.34	
7	0.439 0.426 3.05	1.021 1.013 0.79	1.124 1.130 -0.53	1.297 1.297 0.00	1.022 1.020 0.20	1.289 1.289 0.00	1.017 1.026 -0.88	1.297 1.307 -0.77	1.014 1.026 -1.17	1.270 1.284 -1.09	1.013 1.016 -0.30	1.306 1.295 0.85	1.130 1.129 0.09	1.006 1.010 -0.40	0.422 0.424 -0.47
8	0.413 0.401 2.99	0.927 0.920 0.76	1.031 1.031 0.00	1.135 1.143 -0.70	1.001 1.012 -1.09	1.154 1.165 -0.94	1.299 1.307 -0.61	0.953 0.960 -0.73	1.302 1.307 -0.38	1.159 1.165 -0.52	1.016 1.012 0.40	1.150 1.143 0.61	1.033 1.031 0.19	0.919 0.920 -0.11	0.402 0.401 0.25
9	0.437 0.424 3.07	1.030 1.010 1.98	1.143 1.129 1.24	1.298 1.295 0.23	1.005 1.016 -1.08	1.278 1.284 -0.47	1.031 1.026 0.49	1.306 1.307 -0.08	1.022 1.026 -0.39	1.286 1.289 -0.23	1.014 1.020 -0.59	1.286 1.297 -0.85	1.120 1.130 -0.88	1.012 1.013 -0.10	0.427 0.426 0.23
10		0.917 0.898 2.12	1.261 1.236 2.02	1.223 1.223 0.00	1.297 1.310 -0.99	1.007 1.012 -0.49	1.287 1.289 -0.16	1.160 1.165 -0.43	1.281 1.284 -0.23	1.011 1.012 -0.10	1.311 1.314 -0.23	1.211 1.223 -0.98	1.230 1.236 -0.49	0.906 0.899 0.78	
11		0.438 0.431 1.62	1.080 1.064 1.50	1.265 1.256 0.72	1.028 1.036 -0.77	1.313 1.314 -0.08	1.017 1.020 -0.29	1.009 1.012 -0.30	1.012 1.016 -0.39	1.310 1.310 0.00	1.031 1.036 -0.48	1.248 1.257 -0.72	1.059 1.064 -0.47	0.431 0.431 0.00	
12			0.454 0.449 1.11	0.882 0.881 0.11	1.249 1.257 -0.64	1.219 1.223 -0.33	1.291 1.297 -0.46	1.137 1.143 -0.52	1.295 1.295 0.00	1.226 1.223 0.25	1.257 1.256 0.08	0.876 0.881 -0.57	0.448 0.449 -0.22		
13				0.452 0.449 0.67	1.067 1.064 0.28	1.238 1.236 0.16	1.127 1.130 -0.27	1.031 1.031 0.00	1.131 1.129 0.18	1.243 1.236 0.57	1.067 1.064 0.28	0.448 0.449 -0.22			
14					0.432 0.431 0.23	0.912 0.899 1.45	1.026 1.013 1.28	0.928 0.920 0.87	1.013 1.010 0.30	0.903 0.898 0.56	0.435 0.431 0.93				
15							0.438 0.426 2.82	0.407 0.401 1.50	0.425 0.424 0.24						

Mean Absolute Difference = 0.0065
Standard Deviation = 0.0081

INCORE
ANC
% DIFFERENCE

BURNUP = 11123 MWD/MTU POWER LEVEL = 99.9% D BANK AT 228 STEPS

FIGURE 4.3-15

V.C. SUMMER NUCLEAR STATION CYCLE 3
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-03-037

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.454 0.440 3.18	0.433 0.420 3.10	0.452 0.442 2.26						
2					0.450 0.441 2.04	0.889 0.902 -1.44	1.030 1.025 0.49	0.935 0.930 0.54	1.043 1.028 1.46	0.915 0.903 1.33	0.449 0.441 1.81				
3				0.463 0.461 0.43	1.069 1.065 0.38	1.227 1.244 -1.37	1.108 1.122 -1.25	1.012 1.022 -0.98	1.114 1.123 -0.80	1.241 1.245 -0.32	1.072 1.065 0.66	0.471 0.461 2.17			
4			0.464 0.461 0.65	0.887 0.886 0.11	1.255 1.255 0.00	1.196 1.207 -0.91	1.283 1.296 -1.00	1.118 1.131 -1.15	1.279 1.297 -1.39	1.188 1.207 -1.57	1.254 1.256 -0.16	0.891 0.886 0.56	0.467 0.461 1.30		
5		0.440 0.441 -0.23	1.064 1.065 -0.09	1.254 1.256 -0.16	1.030 1.032 -0.19	1.307 1.309 -0.15	0.996 1.010 -1.39	0.991 1.006 -1.49	0.996 1.013 -1.68	1.296 1.311 -1.14	1.023 1.032 -0.87	1.250 1.255 -0.40	1.077 1.065 1.13	0.450 0.441 2.04	
6		0.894 0.903 -1.00	1.233 1.245 -0.96	1.204 1.207 -0.25	1.319 1.311 0.61	1.008 1.008 0.00	1.267 1.276 -0.71	1.138 1.145 -0.61	1.273 1.280 -0.55	1.002 1.008 -0.60	1.311 1.309 0.15	1.212 1.207 0.41	1.262 1.244 1.45	0.917 0.902 1.66	
7	0.456 0.442 3.17	1.030 1.028 0.19	1.109 1.123 -1.25	1.293 1.297 -0.31	1.017 1.013 0.39	1.282 1.280 0.16	1.005 1.015 -0.99	1.286 1.292 -0.46	1.009 1.015 -0.59	1.273 1.276 -0.24	1.014 1.010 0.40	1.305 1.296 0.69	1.122 1.122 0.00	1.019 1.025 -0.59	0.438 0.440 -0.45
8	0.433 0.420 3.10	0.933 0.930 0.32	1.014 1.022 -0.78	1.120 1.131 -0.97	0.997 1.006 -0.89	1.136 1.145 -0.79	1.287 1.292 -0.39	0.950 0.952 -0.21	1.295 1.292 0.23	1.146 1.145 0.09	1.016 1.006 0.99	1.137 1.131 0.53	1.002 1.002 0.00	0.927 0.930 -0.32	0.421 0.420 0.24
9	0.454 0.440 3.18	1.042 1.025 1.66	1.131 1.122 0.80	1.297 1.296 0.08	1.000 1.010 -0.99	1.269 1.276 -0.55	1.015 1.015 0.00	1.292 1.292 0.00	1.016 1.015 0.10	1.284 1.280 0.31	1.009 1.013 -0.39	1.278 1.297 -1.46	1.108 1.123 -1.34	1.024 1.028 -0.39	0.442 0.442 0.00
10		0.922 0.902 2.22	1.271 1.244 2.17	1.208 1.207 0.08	1.296 1.309 -0.99	1.001 1.008 -0.69	1.282 1.280 0.16	1.145 1.145 0.00	1.281 1.276 0.39	1.011 1.008 0.30	1.310 1.311 -0.08	1.192 1.207 -1.24	1.234 1.245 -0.88	0.909 0.903 0.66	
11		0.449 0.441 1.81	1.084 1.065 1.78	1.266 1.255 0.88	1.023 1.032 -0.87	1.308 1.311 -0.23	1.013 1.013 0.00	1.006 1.006 0.00	1.013 1.010 0.30	1.311 1.309 0.15	1.028 1.032 -0.39	1.247 1.256 -0.72	1.059 1.065 -0.56	0.441 0.441 0.00	
12			0.467 0.461 1.30	0.889 0.886 0.34	1.246 1.256 -0.80	1.201 1.207 -0.50	1.292 1.297 -0.39	1.125 1.131 -0.53	1.300 1.296 0.31	1.216 1.207 0.75	1.261 1.255 0.48	0.885 0.886 -0.11	0.461 0.461 0.00		
13				0.465 0.461 0.87	1.072 1.065 0.66	1.249 1.245 0.32	1.114 1.123 -0.80	1.018 1.022 -0.39	1.124 1.122 0.18	1.258 1.244 1.13	1.075 1.065 0.94	0.461 0.461 0.00			
14					0.444 0.441 0.68	0.917 0.903 1.55	1.033 1.028 0.49	0.933 0.930 0.32	1.027 1.025 0.20	0.911 0.902 1.00	0.449 0.441 1.81				
15							0.453 0.442 2.49	0.426 0.420 1.43	0.441 0.440 0.23						

Mean Absolute Difference = 0.0071
Standard Deviation = 0.0089

INCORE
ANC
% DIFFERENCE

BURNUP = 13581 MWD/MTU   POWER LEVEL = 100.0%   D BANK AT 228 STEPS

FIGURE 4.3-16

V.C. SUMMER NUCLEAR STATION CYCLE 4
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-04-005

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.391 0.393 -0.51	0.384 0.386 -0.52	0.394 0.394 0.00						
2					0.442 0.440 0.45	0.741 0.739 0.27	1.056 1.052 0.38	0.972 0.969 0.31	1.059 1.054 0.47	0.743 0.739 0.54	0.441 0.440 0.23				
3				0.463 0.461 0.43	0.984 0.979 0.51	1.204 1.202 0.17	1.264 1.265 -0.08	1.222 1.225 -0.24	1.260 1.263 -0.24	1.198 1.201 -0.25	0.980 0.982 -0.20	0.463 0.464 -0.22			
4			0.472 0.464 1.72	0.735 0.729 0.82	1.194 1.196 -0.17	1.220 1.228 -0.65	1.230 1.238 -0.65	1.050 1.057 -0.66	1.224 1.233 -0.73	1.215 1.224 -0.74	1.197 1.197 0.00	0.731 0.729 0.27	0.461 0.461 0.00		
5		0.453 0.440 2.95	1.011 0.982 2.95	1.210 1.197 1.09	0.926 0.923 0.33	1.258 1.265 -0.55	1.270 1.292 -1.70	1.178 1.198 -1.67	1.274 1.291 -1.32	1.256 1.264 -0.63	0.926 0.923 0.33	1.204 1.196 0.67	0.981 0.979 0.20	0.434 0.440 -1.36	
6		0.749 0.739 1.35	1.216 1.201 1.25	1.232 1.224 0.65	1.264 1.264 0.00	1.161 1.171 -0.85	1.193 1.215 -1.81	1.200 1.224 -1.96	1.198 1.217 -1.56	1.157 1.171 -1.20	1.271 1.265 0.47	1.239 1.228 0.90	1.218 1.202 1.33	0.748 0.739 1.22	
7	0.389 0.394 -1.27	1.054 1.054 0.00	1.267 1.263 0.32	1.227 1.233 -0.49	1.277 1.291 -1.08	1.199 1.217 -1.48	0.977 0.995 -1.81	1.182 1.208 -2.15	0.978 0.995 -1.71	1.194 1.215 -1.73	1.290 1.292 -0.15	1.256 1.238 1.45	1.287 1.265 1.74	1.073 1.052 2.00	0.401 0.393 2.04
8	0.381 0.386 -1.30	0.968 0.969 -0.10	1.228 1.225 0.24	1.050 1.057 -0.66	1.178 1.198 -1.67	1.202 1.224 -1.80	1.183 1.208 -2.07	0.925 0.948 -2.43	1.193 1.208 -1.24	1.210 1.224 -1.14	1.206 1.198 0.67	1.073 1.057 1.51	1.243 1.225 1.47	0.985 0.969 1.65	0.393 0.386 1.81
9	0.389 0.393 -1.02	1.059 1.052 0.67	1.284 1.265 1.50	1.236 1.238 -0.16	1.273 1.292 -1.47	1.207 1.215 -0.66	1.011 0.995 1.61	1.202 1.208 -0.50	0.988 0.995 -0.70	1.210 1.217 -0.58	1.283 1.291 -0.62	1.225 1.233 -0.65	1.264 1.263 0.08	1.067 1.054 1.23	0.398 0.394 1.02
10		0.753 0.739 1.89	1.223 1.202 1.75	1.233 1.228 0.41	1.261 1.265 -0.32	1.175 1.171 0.34	1.211 1.217 -0.49	1.208 1.224 -1.31	1.210 1.215 -0.41	1.173 1.171 0.17	1.265 1.264 0.08	1.200 1.224 -1.96	1.192 1.201 -0.75	0.750 0.739 1.49	
11		0.444 0.440 0.91	0.990 0.979 1.12	1.204 1.196 0.67	0.924 0.923 0.11	1.271 1.264 0.55	1.286 1.291 -0.39	1.192 1.198 -0.50	1.287 1.292 -0.39	1.271 1.265 0.47	0.921 0.923 -0.22	1.182 1.197 -1.25	0.975 0.982 -0.71	0.440 0.440 0.00	
12			0.462 0.461 0.22	0.729 0.729 0.00	1.195 1.197 -0.17	1.224 1.224 0.00	1.224 1.233 -0.73	1.049 1.057 0.76	1.243 1.238 0.40	1.241 1.228 1.06	1.204 1.196 0.67	0.723 0.729 -0.82	0.460 0.464 -0.86		
13				0.464 0.464 0.00	0.983 0.982 0.10	1.203 1.201 0.17	1.270 1.263 0.55	1.239 1.225 1.14	1.290 1.265 1.98	1.232 1.202 2.50	0.988 0.979 0.98	0.457 0.461 -0.87			
14					0.440 0.440 0.00	0.740 0.739 0.14	1.060 1.054 0.57	0.982 0.969 1.34	1.075 1.052 2.19	0.759 0.739 2.71	0.449 0.440 2.05				
15							0.395 0.394 0.25	0.392 0.386 1.55	0.406 0.393 3.31						

Mean Absolute Difference = 0.0088
Standard Deviation = 0.0115

INCORE
ANC
% DIFFERENCE

BURNUP = 1232 MWD/MTU POWER LEVEL = 99.8% D BANK AT 228 STEPS

FIGURE 4.3-17

V.C. SUMMER NUCLEAR STATION CYCLE 4
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-04-010

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.384 0.386 -0.52	0.378 0.381 -0.79	0.388 0.387 0.26						
2					0.445 0.441 0.91	0.724 0.723 0.14	1.026 1.024 0.20	0.936 0.936 0.00	1.031 1.026 0.49	0.732 0.723 1.24	0.445 0.441 0.91				
3				0.468 0.464 0.86	0.996 0.986 1.01	1.207 1.206 0.08	1.219 1.224 -0.41	1.240 1.246 -0.48	1.218 1.222 -0.33	1.205 1.205 0.00	0.991 0.988 0.30	0.470 0.467 0.64			
4			0.472 0.467 1.07	0.730 0.726 0.55	1.201 1.199 0.17	1.196 1.201 -0.42	1.265 1.272 -0.55	1.033 1.040 -0.67	1.255 1.267 -0.95	1.185 1.198 -1.09	1.202 1.200 0.17	0.731 0.726 0.69	0.467 0.464 0.65		
5		0.447 0.441 1.36	1.002 0.988 1.42	1.206 1.200 0.50	0.915 0.913 0.22	1.297 1.301 -0.31	1.250 1.266 -1.26	1.151 1.167 -1.37	1.252 1.265 -1.03	1.295 1.300 -0.38	0.918 0.913 0.55	1.208 1.199 0.75	0.992 0.986 0.61	0.438 0.441 -0.68	
6		0.726 0.723 0.41	1.210 1.205 0.41	1.199 1.198 0.08	1.299 1.300 -0.08	1.162 1.170 -0.68	1.257 1.272 -1.18	1.196 1.214 -1.48	1.262 1.274 -0.94	1.162 1.170 -0.68	1.310 1.301 0.69	1.210 1.201 0.75	1.222 1.206 1.33	0.732 0.723 1.24	
7	0.384 0.387 -0.78	1.023 1.026 -0.29	1.220 1.222 -0.16	1.260 1.267 -0.55	1.254 1.265 -0.87	1.259 1.274 -1.18	0.999 1.015 -1.58	1.247 1.270 -1.81	1.002 1.015 -1.28	1.259 1.272 -1.02	1.268 1.266 0.16	1.286 1.272 1.10	1.242 1.224 1.47	1.043 1.024 1.86	0.393 0.386 1.81
8	0.377 0.381 -1.05	0.933 0.936 -0.32	1.244 1.246 -0.16	1.033 1.040 -0.67	1.153 1.167 -1.20	1.198 1.214 -1.32	1.248 1.270 -1.73	0.952 0.974 -2.26	1.258 1.270 -0.94	1.203 1.214 -0.91	1.174 1.167 0.60	1.051 1.040 1.06	1.263 1.246 1.36	0.952 0.936 1.71	0.388 0.381 1.84
9	0.383 0.386 -0.78	1.027 1.024 0.29	1.232 1.224 0.35	1.269 1.272 -0.24	1.253 1.266 -1.03	1.269 1.272 -0.24	1.031 1.015 1.58	1.266 1.270 -0.31	1.009 1.015 -0.59	1.271 1.274 -0.24	1.259 1.265 -0.47	1.257 1.267 -0.79	1.221 1.222 -0.08	1.038 1.026 1.17	0.391 0.387 1.03
10		0.730 0.723 0.97	1.218 1.206 1.00	1.204 1.201 0.25	1.299 1.301 -0.15	1.176 1.170 0.51	1.275 1.274 0.08	1.205 1.214 -0.74	1.270 1.272 -0.16	1.173 1.170 0.26	1.301 1.300 0.08	1.176 1.198 -1.84	1.191 1.205 -1.16	0.732 0.723 1.24	
11		0.444 0.441 0.68	0.993 0.986 0.71	1.204 1.199 0.42	0.915 0.913 0.22	1.307 1.300 0.54	1.264 1.265 -0.08	1.164 1.167 -0.26	1.262 1.266 -0.32	1.306 1.301 0.38	0.910 0.913 -0.33	1.185 1.200 -1.25	0.978 0.988 -1.01	0.439 0.441 -0.45	
12			0.466 0.464 0.43	0.727 0.726 0.14	1.199 1.200 -0.08	1.196 1.198 -0.17	1.260 1.267 -0.55	1.034 1.040 -0.58	1.276 1.272 0.31	1.211 1.201 0.83	1.204 1.199 0.42	0.720 0.726 -0.83	0.462 0.467 -1.07		
13				0.470 0.467 0.64	0.997 0.988 0.91	1.211 1.205 0.50	1.224 1.222 0.16	1.255 1.246 0.72	1.241 1.224 1.39	1.230 1.206 1.99	0.991 0.986 0.51	0.460 0.464 -0.86			
14					0.445 0.441 0.91	0.728 0.723 0.69	1.031 1.026 0.49	0.946 0.936 1.07	1.040 1.024 1.56	0.737 0.723 1.94	0.448 0.441 1.59				
15							0.389 0.387 0.52	0.386 0.381 1.31	0.395 0.386 2.33						

Mean Absolute Difference = 0.0073
Standard Deviation = 0.0092

INCORE ANC % DIFFERENCE

BURNUP = 4154 MWD/MTU   POWER LEVEL = 100.0%   D BANK AT 228 STEPS

FIGURE 4.3-18

V.C. SUMMER NUCLEAR STATION CYCLE 4
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-04-012

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.385 0.386 -0.26	0.381 0.383 -0.52	0.389 0.387 0.52						
2					0.446 0.443 0.68	0.717 0.716 0.14	1.013 1.008 0.50	0.925 0.922 0.33	1.018 1.010 0.79	0.724 0.716 1.12	0.448 0.443 1.13				
3				0.471 0.468 0.64	0.994 0.987 0.71	1.205 1.203 0.17	1.196 1.201 -0.42	1.255 1.259 -0.32	1.195 1.199 -0.33	1.203 1.203 0.00	0.993 0.989 0.40	0.475 0.471 0.85			
4			0.474 0.471 0.64	0.729 0.726 0.41	1.200 1.194 0.50	1.179 1.183 -0.34	1.282 1.291 -0.70	1.026 1.034 -0.77	1.275 1.286 -0.86	1.172 1.180 -0.68	1.200 1.195 0.42	0.732 0.726 0.83	0.472 0.468 0.85		
5		0.447 0.443 0.90	0.998 0.989 0.91	1.202 1.195 0.59	0.911 0.906 0.55	1.304 1.307 -0.23	1.233 1.251 -1.44	1.134 1.151 -1.48	1.237 1.250 -1.04	1.314 1.316 -0.15	0.911 0.906 0.55	1.205 1.194 0.92	0.995 0.987 0.81	0.442 0.443 -0.23	
6		0.717 0.716 0.14	1.205 1.203 0.17	1.182 1.180 0.17	1.320 1.316 0.30	1.162 1.168 -0.51	1.291 1.306 -1.15	1.194 1.211 -1.40	1.297 1.308 -0.84	1.163 1.168 -0.43	1.326 1.317 0.68	1.192 1.183 0.76	1.218 1.203 1.25	0.724 0.716 1.12	
7	0.384 0.387 -0.78	1.009 1.010 -0.10	1.201 1.199 0.17	1.280 1.286 -0.47	1.236 1.250 -1.12	1.291 1.308 -1.30	1.015 1.028 -1.26	1.288 1.307 -1.45	1.018 1.028 -0.97	1.294 1.306 -0.92	1.252 1.251 0.08	1.304 1.291 1.01	1.215 1.201 1.17	1.023 1.008 1.49	0.392 0.386 1.55
8	0.379 0.383 -1.04	0.920 0.922 -0.22	1.262 1.259 0.24	1.027 1.034 -0.68	1.136 1.151 -1.30	1.193 1.211 -1.49	1.287 1.307 -1.53	0.975 0.992 -1.71	1.298 1.307 -0.69	1.202 1.211 -0.74	1.156 1.151 0.43	1.044 1.034 0.97	1.274 1.259 1.19	0.934 0.922 1.30	0.388 0.383 1.31
9	0.383 0.386 -0.78	1.010 1.008 0.20	1.208 1.201 0.58	1.286 1.291 -0.39	1.237 1.251 -1.12	1.304 1.306 -0.15	1.049 1.028 2.04	1.313 1.307 0.46	1.025 1.028 -0.29	1.306 1.308 -0.15	1.239 1.250 -0.88	1.272 1.286 -1.09	1.195 1.199 -0.33	1.020 1.010 0.99	0.390 0.387 0.78
10		0.718 0.716 0.28	1.208 1.203 0.42	1.182 1.183 -0.08	1.313 1.317 -0.30	1.175 1.168 0.60	1.311 1.308 0.23	1.206 1.211 -0.41	1.303 1.306 -0.23	1.171 1.168 0.26	1.311 1.316 -0.38	1.154 1.180 -2.20	1.186 1.203 -1.41	0.723 0.716 0.98	
11		0.446 0.443 0.68	0.994 0.987 0.71	1.197 1.194 0.25	0.903 0.906 -0.33	1.325 1.316 0.68	1.252 1.250 0.16	1.151 1.151 0.00	1.244 1.251 -0.56	1.319 1.317 0.15	0.902 0.906 -0.44	1.182 1.195 -1.09	0.981 0.989 -0.81	0.444 0.443 0.23	
12			0.472 0.468 0.85	0.727 0.726 0.14	1.189 1.195 -0.50	1.179 1.180 -0.08	1.282 1.286 -0.31	1.029 1.034 -0.48	1.290 1.291 -0.08	1.186 1.183 0.25	1.196 1.194 0.17	0.722 0.726 -0.55	0.469 0.471 -0.42		
13				0.475 0.471 0.85	0.999 0.989 1.01	1.212 1.203 0.75	1.205 1.199 0.50	1.271 1.259 0.95	1.213 1.201 1.00	1.219 1.203 1.33	0.989 0.987 0.20	0.466 0.468 -0.43			
14					0.448 0.443 1.13	0.723 0.716 0.98	1.018 1.010 0.79	0.933 0.922 1.19	1.020 1.008 1.19	0.726 0.716 1.40	0.446 0.443 0.68				
15							0.391 0.387 1.03	0.389 0.383 1.57	0.395 0.386 2.33						

Mean Absolute Difference = 0.0069
Standard Deviation = 0.0086

INCORE
ANC
% DIFFERENCE

BURNUP = 6398 MWD/MTU POWER LEVEL = 99.8% D BANK AT 228 STEPS

FIGURE 4.3-19

V.C. SUMMER NUCLEAR STATION CYCLE 4
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-04-016

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.390 0.390 0.00	0.388 0.389 -0.26	0.393 0.391 0.51						
2					0.452 0.449 0.67	0.710 0.713 -0.42	1.002 0.998 0.40	0.918 0.915 0.33	1.010 1.000 1.00	0.724 0.713 1.54	0.455 0.449 1.34				
3				0.479 0.475 0.84	0.999 0.990 0.91	1.195 1.201 -0.50	1.175 1.183 -0.68	1.264 1.271 -0.55	1.182 1.181 0.08	1.219 1.200 1.58	1.008 0.992 1.61	0.483 0.477 1.26			
4			0.482 0.477 1.05	0.732 0.728 0.55	1.194 1.190 0.34	1.159 1.167 -0.69	1.292 1.303 -0.84	1.019 1.029 -0.97	1.289 1.299 -0.77	1.159 1.164 -0.43	1.196 1.192 0.34	0.731 0.728 0.41	0.478 0.475 0.63		
5		0.453 0.449 0.89	1.003 0.992 1.11	1.199 1.192 0.59	0.905 0.901 0.44	1.323 1.325 -0.15	1.215 1.236 -1.70	1.118 1.138 -1.76	1.218 1.235 -1.38	1.321 1.325 -0.30	0.904 0.901 0.33	1.195 1.190 0.42	0.996 0.990 0.61	0.449 0.449 0.00	
6		0.716 0.713 0.42	1.205 1.200 0.42	1.168 1.164 0.34	1.330 1.325 0.38	1.156 1.163 -0.60	1.311 1.328 -1.28	1.184 1.203 -1.58	1.313 1.330 -1.28	1.151 1.163 -1.03	1.327 1.325 0.15	1.170 1.167 0.26	1.217 1.201 1.33	0.724 0.713 1.54	
7	0.389 0.391 -0.51	1.005 1.000 0.50	1.186 1.181 0.42	1.298 1.299 -0.08	1.225 1.235 -0.81	1.315 1.330 -1.13	1.021 1.035 -1.35	1.309 1.332 -1.73	1.021 1.035 -1.35	1.310 1.328 -1.36	1.227 1.236 -0.73	1.322 1.303 1.46	1.197 1.183 1.18	1.011 0.998 1.30	0.394 0.390 1.03
8	0.394 0.389 1.29	0.920 0.915 0.55	1.278 1.271 0.55	1.024 1.029 -0.49	1.122 1.138 -1.41	1.185 1.203 -1.50	1.308 1.332 -1.80	0.982 1.003 -2.09	1.321 1.332 -0.83	1.193 1.203 -0.83	1.143 1.138 0.44	1.043 1.029 1.36	1.286 1.271 1.18	0.925 0.915 1.09	0.392 0.389 0.77
9	0.397 0.390 1.79	1.013 0.998 1.50	1.200 1.183 1.44	1.304 1.303 0.08	1.221 1.236 -1.21	1.328 1.328 0.00	1.060 1.035 2.42	1.338 1.332 0.45	1.032 1.035 -0.29	1.329 1.330 -0.08	1.226 1.235 -0.73	1.286 1.299 -1.00	1.171 1.181 -0.85	1.003 1.000 0.30	0.388 0.391 -0.77
10		0.724 0.713 1.54	1.219 1.201 1.50	1.169 1.167 0.17	1.320 1.325 -0.38	1.170 1.163 0.60	1.333 1.330 0.23	1.197 1.203 -0.50	1.319 1.328 -0.68	1.161 1.163 -0.17	1.314 1.325 -0.83	1.135 1.164 -2.49	1.161 1.200 -3.25	0.689 0.713 -3.37	
11		0.455 0.449 1.34	1.005 0.990 1.52	1.199 1.190 0.76	0.895 0.901 -0.67	1.336 1.325 0.83	1.238 1.235 0.24	1.138 1.138 0.00	1.204 1.236 -2.59	1.307 1.325 -1.36	0.888 0.901 -1.44	1.176 1.192 -1.34	0.980 0.992 -1.21	0.448 0.449 -0.22	
12			0.481 0.475 1.26	0.730 0.728 0.27	1.182 1.192 -0.84	1.164 1.164 0.00	1.291 1.299 -0.62	1.020 1.029 -0.87	1.296 1.303 -0.54	1.168 1.167 0.09	1.189 1.190 -0.08	0.724 0.728 -0.55	0.476 0.477 -0.21		
13				0.483 0.477 1.26	1.004 0.992 1.21	1.213 1.200 1.08	1.192 1.181 0.93	1.293 1.271 1.73	1.206 1.183 1.94	1.224 1.201 1.92	0.996 0.990 0.61	0.473 0.475 -0.42			
14					0.454 0.449 1.11	0.721 0.713 1.12	1.011 1.000 1.10	0.938 0.915 2.51	[illegible] 0.998 2.10	0.728 0.713 2.10	0.456 0.449 1.56				
15							0.395 0.391 1.02	0.398 0.389 2.31	0.400 0.390 2.56						

Mean Absolute Difference = 0.0093
Standard Deviation = 0.0118

INCORE
ANC
% DIFFERENCE

BURNUP = 8608 MWD/MTU   POWER LEVEL = 100.1%   D BANK AT 228 STEPS

FIGURE 4.3-20

V.C. SUMMER NUCLEAR STATION CYCLE 4
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-04-021

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.409 0.407 0.49	0.411 0.409 0.49	0.413 0.408 1.23						
2					0.475 0.465 2.15	0.720 0.722 -0.28	1.008 1.000 0.80	0.928 0.921 0.76	1.018 1.002 1.60	0.739 0.723 2.21	0.476 0.465 2.37				
3				0.504 0.494 2.02	1.026 1.005 2.09	1.198 1.202 -0.33	1.161 1.165 -0.34	1.283 1.285 -0.16	1.170 1.164 0.52	1.230 1.202 2.33	1.031 1.007 2.38	0.501 0.496 1.01			
4			0.504 0.496 1.61	0.750 0.745 0.67	1.199 1.191 0.67	1.135 1.147 -1.05	1.296 1.306 -0.77	1.013 1.022 -0.88	1.298 1.303 -0.38	1.134 1.145 -0.96	1.193 1.193 0.00	0.745 0.745 0.00	0.500 0.494 1.21		
5		0.470 0.465 1.08	1.018 1.007 1.09	1.197 1.193 0.34	0.898 0.897 0.11	1.311 1.319 -0.61	1.181 1.206 -2.07	1.091 1.114 -2.06	1.184 1.205 -1.74	1.307 1.319 -0.91	0.896 0.897 -0.11	1.193 1.191 0.17	1.018 1.005 1.29	0.472 0.465 1.51	
6		0.725 0.723 0.28	1.207 1.202 0.42	1.147 1.145 0.17	1.320 1.319 0.08	1.132 1.142 -0.88	1.306 1.328 -1.66	1.156 1.179 -1.95	1.309 1.330 -1.58	1.126 1.142 -1.40	1.315 1.319 -0.30	1.147 1.147 0.00	1.226 1.202 2.00	0.740 0.722 2.49	
7	0.407 0.408 -0.25	1.009 1.002 0.70	1.171 1.164 0.60	1.299 1.303 -0.31	1.189 1.205 -1.33	1.311 1.330 -1.43	1.013 1.031 -1.75	1.312 1.337 -1.87	1.014 1.031 -1.65	1.309 1.328 -1.43	1.194 1.206 -1.00	1.324 1.306 1.38	1.182 1.165 1.46	1.017 1.000 1.70	0.413 0.407 1.47
8	0.415 0.409 1.47	0.927 0.921 0.65	1.295 1.285 0.78	1.015 1.022 -0.68	1.092 1.114 -1.97	1.156 1.179 -1.95	1.313 1.337 -1.80	0.982 1.004 -2.19	1.325 1.337 -0.90	1.168 1.179 -0.93	1.118 1.114 0.36	1.035 1.022 1.27	1.304 1.285 1.48	0.933 0.921 1.30	0.415 0.409 1.47
9	0.414 0.407 1.72	1.018 1.000 1.80	1.184 1.165 1.63	1.305 1.306 -0.08	1.184 1.206 -1.82	1.323 1.328 -0.38	1.054 1.031 2.23	1.341 1.337 0.30	1.026 1.031 -0.48	1.328 1.330 -0.15	1.194 1.205 -0.91	1.287 1.303 -1.23	1.153 1.164 -0.95	1.007 1.002 0.50	0.406 0.408 -0.49
10		0.735 0.722 1.80	1.224 1.202 1.83	1.141 1.147 -0.52	1.298 1.319 -1.59	1.140 1.142 -0.18	1.331 1.330 0.08	1.169 1.179 -0.85	1.316 1.328 -0.90	1.138 1.142 -0.35	1.304 1.319 -1.14	1.113 1.145 -2.79	1.157 1.202 -3.74	0.695 0.723 -3.87	
11		0.474 0.465 1.94	1.024 1.005 1.89	1.195 1.191 0.34	0.874 0.897 -2.56	1.322 1.319 0.23	1.208 1.205 0.25	1.113 1.114 -0.09	1.171 1.206 -2.90	1.300 1.319 -1.44	0.883 0.897 -1.56	1.180 1.193 -1.09	0.999 1.007 -0.79	0.470 0.465 1.08	
12			0.503 0.494 1.82	0.742 0.745 -0.40	1.160 1.193 -2.77	1.136 1.145 -0.79	1.295 1.303 -0.61	1.013 1.022 -0.88	1.295 1.306 -0.84	1.145 1.147 -0.17	1.191 1.191 0.00	0.744 0.745 -0.13	0.499 0.496 0.60		
13				0.507 0.496 2.22	1.031 1.007 2.38	1.223 1.202 1.75	1.178 1.164 1.20	1.314 1.285 2.26	1.190 1.165 2.15	1.227 1.202 2.08	1.012 1.005 0.70	0.497 0.494 0.61			
14					0.476 0.465 2.37	0.739 0.723 2.21	1.025 1.002 2.30	0.950 0.921 3.15	1.024 1.000 2.40	0.738 0.722 2.22	0.471 0.465 1.29				
15							0.417 0.408 2.21	0.422 0.409 3.18	0.420 0.407 3.19						

Mean Absolute Difference = 0.0119
Standard Deviation = 0.0148

INCORE ANC % DIFFERENCE

BURNUP = 12192 MWD/MTU   POWER LEVEL = 99.9%   D BANK AT 226 STEPS

FIGURE 4.3-21

V.C. SUMMER NUCLEAR STATION CYCLE 4
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-04-026

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.434 0.428 1.40	0.439 0.434 1.15	0.436 0.429 1.63						
2					0.496 0.484 2.48	0.733 0.738 -0.68	1.021 1.014 0.69	0.943 0.937 0.64	1.033 1.015 1.77	0.755 0.738 2.30	0.495 0.484 2.27				
3				0.528 0.515 2.52	1.048 1.022 2.54	1.194 1.203 -0.75	1.150 1.157 -0.61	1.280 1.287 -0.54	1.161 1.157 0.35	1.231 1.203 2.33	1.048 1.024 2.34	0.518 0.518 0.00			
4			0.528 1.518 1.93	0.776 0.766 1.31	1.210 1.195 1.26	1.125 1.135 -0.88	1.280 1.293 -1.01	1.006 1.017 -1.08	1.283 1.29[illegible] -0.62	1.133 1.134 -0.09	1.196 1.196 0.00	0.765 0.766 -0.13	0.522 0.515 1.36		
5		0.492 0.484 1.65	1.041 1.024 1.66	1.204 1.196 0.67	0.901 0.899 0.22	1.293 1.301 -0.61	1.154 1.182 -2.37	1.072 1.098 -2.37	1.156 1.181 -2.12	1.293 1.301 -0.61	0.897 0.899 -0.22	1.195 1.195 0.00	1.035 1.022 1.27	0.496 0.484 2.48	
6		0.743 0.738 0.68	1.210 1.203 0.58	1.138 1.134 0.35	1.303 1.301 0.15	1.112 1.122 -0.89	1.279 1.305 -1.99	1.130 1.155 -2.16	1.281 1.307 -1.99	1.108 1.122 -1.25	1.296 1.301 -0.38	1.135 1.135 0.00	1.228 1.203 2.08	0.761 0.738 3.12	
7	[illegible] [illegible] -0.47	1.022 1.015 0.69	1.163 1.157 0.52	1.286 1.291 -0.39	1.162 1.181 -1.61	1.286 1.307 -1.61	0.998 1.019 -2.06	1.288 1.317 -2.20	1.000 1.019 -1.86	1.285 1.305 -1.53	1.168 1.182 -1.18	1.300 1.293 0.54	1.171 1.157 1.21	1.027 1.014 1.28	0.435 0.428 1.64
8	0.441 0.434 1.61	0.943 0.937 0.64	1.296 1.287 0.70	1.011 1.017 -0.59	1.077 1.098 -1.91	1.134 1.155 -1.82	1.294 1.317 -1.75	0.973 0.995 -2.21	1.304 1.317 -0.99	1.144 1.155 -0.95	1.102 1.098 0.36	1.022 1.017 0.49	1.300 1.287 1.01	0.946 0.937 0.96	0.439 0.434 1.15
9	0.435 0.428 1.64	1.031 1.014 1.68	1.177 1.157 1.73	1.293 1.293 0.00	1.162 1.182 -1.69	1.299 1.305 -0.46	1.038 1.019 1.86	1.315 1.317 -0.15	1.013 1.019 -0.59	1.303 1.307 -0.31	1.171 1.181 -0.85	1.270 1.291 -1.63	1.142 1.157 -1.30	1.014 1.015 -0.10	0.425 0.429 -[illegible]93
10		0.750 0.738 1.63	1.222 1.203 1.58	1.133 1.135 -0.18	1.286 1.301 -1.15	1.121 1.122 -0.09	1.305 1.307 -0.15	1.144 1.155 -0.95	1.293 1.305 -0.92	1.116 1.122 -0.53	1.287 1.301 -1.08	1.103 1.134 -2.73	1.159 1.203 -3.66	0.711 0.738 -3.66	
11		0.494 0.484 2.07	1.042 1.022 1.96	1.201 1.195 0.50	0.881 0.899 -2.00	1.305 1.301 0.31	1.183 1.181 0.17	1.097 1.098 -0.09	1.150 1.182 -2.71	1.281 1.301 -1.54	0.887 0.899 -1.33	1.188 1.196 -0.67	1.021 1.024 -0.29	0.495 0.484 2.27	
12			0.527 0.515 2.33	0.766 0.766 0.00	1.170 1.196 -2.17	1.128 1.134 -0.53	1.284 1.291 -0.54	1.008 1.017 -0.88	1.282 1.293 -0.85	1.132 1.135 -0.26	1.195 1.195 0.00	0.769 0.766 0.39	0.525 0.518 1.35		
13				0.530 0.518 2.32	1.049 1.024 2.44	1.225 1.203 1.83	1.169 1.157 1.04	1.314 1.287 2.10	1.180 1.157 1.99	1.226 1.203 1.91	1.033 1.022 0.78	0.523 0.515 1.55			
14					0.496 0.484 2.48	0.756 0.738 2.44	1.040 1.015 2.46	0.965 0.937 2.99	1.036 1.014 2.17	0.753 0.738 2.03	0.489 0.484 1.03				
15							0.439 0.429 2.33	0.447 0.434 3.00	0.441 0.428 3.04						

Mean Absolute Difference = 0.0121
Standard Deviation = 0.0149

INCORE
ANC
% DIFFERENCE

BURNUP = 14970 MWD/MTU   POWER LEVEL = 100.0%   D BANK AT 228 STEPS

FIGURE 4.3-22

V.C. SUMMER NUCLEAR STATION CYCLE 5
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-05-006

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.368 0.372 -1.08	0.407 0.412 -1.21	0.375 0.371 1.08						
2					0.444 0.459 -3.27	0.976 0.979 -0.31	1.047 1.044 0.29	0.873 0.872 0.11	1.061 1.043 1.73	1.016 0.980 3.67	0.470 0.459 2.40				
3				0.398 0.407 -2.21	1.110 1.128 -1.60	1.222 1.227 -0.41	1.146 1.149 -0.26	1.303 1.301 0.15	1.156 1.150 0.52	1.245 1.229 1.30	1.148 1.132 1.141	0.414 0.409 1.22			
4			0.406 0.409 -0.73	0.718 0.727 -1.24	1.108 1.127 -1.69	1.100 1.112 -1.08	1.114 1.118 -0.36	1.100 1.105 -0.45	1.113 1.119 -0.54	1.106 1.115 -0.81	1.142 1.132 0.88	0.737 0.727 1.38	0.412 1.407 1.23		
5		0.467 0.459 1.74	1.155 1.132 2.03	1.133 1.132 0.09	1.049 1.057 -0.76	1.246 1.253 -0.56	1.103 1.124 -1.87	1.068 1.090 -2.02	1.100 1.125 -2.22	1.240 1.255 -1.20	1.055 1.057 -0.19	1.147 1.127 1.77	1.148 1.128 1.77	0.458 1.459 -0.22	
6		0.992 0.980 1.22	1.244 1.229 1.22	1.116 1.115 0.09	1.250 1.255 -0.40	1.173 1.186 -1.10	1.284 1.309 -1.91	1.100 1.129 -2.57	1.271 1.309 -2.90	1.160 1.186 -2.19	1.260 1.253 0.56	1.128 1.112 1.44	1.254 1.227 2.20	0.997 0.979 1.84	
7	0.360 0.371 -2.96	1.042 1.043 -0.10	1.159 1.150 0.78	1.120 1.119 0.09	1.118 1.125 -0.62	1.296 1.309 -0.99	1.128 1.159 -2.67	1.278 1.314 -2.74	1.124 1.159 -3.02	1.273 1.309 -2.75	1.110 1.124 -1.25	1.145 1.118 2.42	1.177 1.149 2.44	1.076 1.044 3.07	0.381 0.372 2.42
8	0.399 0.412 -3.16	0.868 0.872 -0.46	1.317 1.301 1.23	1.097 1.105 -0.72	1.066 1.090 -2.20	1.104 1.129 -2.21	1.287 1.314 -2.05	1.022 1.054 -3.04	1.287 1.314 -2.05	1.103 1.129 -2.30	1.086 1.090 -0.37	1.133 1.105 2.53	1.337 1.301 2.77	0.893 0.872 2.41	0.420 0.412 1.94
9	0.361 0.372 -2.96	1.046 1.044 0.19	1.165 1.149 1.39	1.113 1.118 -0.45	1.099 1.124 -2.22	1.299 1.309 -0.76	1.176 1.159 1.47	1.306 1.314 -0.61	1.138 1.159 -1.81	1.290 1.309 -1.45	1.116 1.125 -0.80	1.138 1.119 1.70	1.176 1.150 2.26	1.075 1.043 3.07	0.380 0.371 2.43
10		0.993 0.979 1.43	1.245 1.227 1.47	1.099 1.112 -1.17	1.225 1.253 -2.23	1.173 1.186 -1.10	1.316 1.309 0.53	1.119 1.129 -0.89	1.307 1.309 -0.15	1.184 1.186 -0.17	1.264 1.255 0.72	1.112 1.115 -0.27	1.258 1.229 2.36	1.013 0.980 3.37	
11		0.457 0.459 -0.44	1.126 1.128 -0.18	1.116 1.127 -0.98	1.028 1.057 -2.74	1.252 1.255 -0.24	1.126 1.125 0.09	1.090 1.090 0.00	1.130 1.124 0.53	1.268 1.253 1.20	1.059 1.057 0.19	1.133 1.132 0.09	1.147 1.132 1.33	0.470 0.459 2.40	
12			0.398 0.407 -2.21	0.709 0.727 -2.48	1.104 1.132 -2.47	1.100 1.115 -1.35	1.123 1.119 0.36	1.109 1.105 0.36	1.134 1.118 1.43	1.127 1.112 1.35	1.139 1.127 1.06	0.725 0.727 -0.28	0.409 0.409 0.00		
13				0.403 0.409 -1.47	1.127 1.132 -0.44	1.228 1.229 -0.08	1.163 1.150 1.13	1.323 1.301 1.69	1.172 1.149 2.00	1.255 1.227 2.28	1.140 1.128 1.06	0.407 0.407 0.00			
14					0.457 0.459 -0.44	0.978 0.980 -0.20	1.056 1.043 1.25	0.883 0.872 1.25	1.068 1.044 2.30	1.001 0.979 2.25	0.468 0.459 1.96				
15							0.370 0.371 -0.27	0.415 0.412 0.73	0.379 0.372 1.88						

Legend: INCORE / ANC / % DIFFERENCE

Mean Absolute Difference = 0.0134
Standard Deviation = 0.0168

BURNUP = 423 MWD/MTU   POWER LEVEL = 99.9%   D BANK AT 230 STEPS

FIGURE 4.3-23

V.C. SUMMER NUCLEAR STATION CYCLE 5
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-05-012

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.372 0.380 -2.11	0.411 0.420 -2.14	0.377 0.379 -0.53						
2					0.450 0.459 -1.96	0.963 0.967 -0.41	1.059 1.057 0.19	0.862 0.863 -0.12	1.066 1.056 0.95	0.985 0.967 1.86	0.464 0.459 1.09				
3				0.414 0.418 -0.96	1.119 1.124 -0.44	1.262 1.259 0.24	1.111 1.113 -0.18	1.289 1.285 0.31	1.114 1.114 0.00	1.265 1.261 0.32	1.133 1.127 0.53	0.421 0.420 0.24			
4			0.415 0.420 -1.19	0.744 0.752 -1.06	1.183 1.192 -0.76	1.096 1.105 -0.81	1.066 1.071 -0.47	1.041 1.047 -0.57	1.065 1.071 -0.56	1.092 1.107 -1.36	1.200 1.196 0.33	0.755 0.752 0.40	0.420 0.418 0.48		
5		0.459 0.459 0.00	1.132 1.127 0.44	1.202 1.196 0.50	1.079 1.077 0.19	1.305 1.301 0.31	1.067 1.088 -1.93	1.020 1.040 -1.92	1.066 1.089 -2.11	1.286 1.303 -1.30	1.070 1.077 -0.65	1.205 1.192 1.09	1.136 1.124 1.07	0.455 0.459 -0.87	
6		0.968 0.967 0.10	1.260 1.261 -0.08	1.105 1.107 -0.18	1.313 1.303 0.77	1.172 1.176 -0.34	1.311 1.327 -1.21	1.[illegible]35 1.1[illegible]3 -1.[illegible]3	1.303 1.327 -1.81	1.156 1.176 -1.70	1.302 1.301 0.08	1.110 1.105 0.45	1.270 1.259 0.87	0.962 0.967 -0.52	
7	0.373 0.379 -1.58	1.060 1.056 0.38	1.122 1.114 0.72	1.078 1.071 0.65	1.095 1.089 0.55	1.331 1.327 0.30	1.127 1.146 -1.66	1.325 1.342 -1.27	1.121 1.146 -2.18	1.303 1.327 -1.81	1.072 1.088 -1.47	1.087 1.071 1.49	1.123 1.113 0.90	1.073 1.057 1.51	0.376 0.380 -1.05
8	0.413 0.420 -1.67	0.863 0.863 0.00	1.303 1.285 1.40	1.040 1.047 -0.67	1.019 1.040 -2.02	1.083 1.103 -1.81	1.334 1.342 -0.60	1.038 1.053 -1.42	1.330 1.342 -0.89	1.087 1.103 -1.45	1.033 1.040 -0.67	1.064 1.047 1.62	1.302 1.285 1.32	0.870 0.863 0.81	0.419 0.420 -0.24
9	0.374 0.380 -1.58	1.072 1.057 1.42	1.139 1.113 2.34	1.071 1.071 0.60	1.064 1.088 -2.21	1.324 1.327 -0.23	1.178 1.146 2.79	1.353 1.342 0.82	1.131 1.146 -1.31	1.317 1.327 -0.75	1.077 1.089 -1.10	1.081 1.071 0.93	1.126 1.114 1.08	1.075 1.056 1.80	0.382 0.379 0.79
10		0.992 0.967 2.59	1.291 1.259 2.54	1.093 1.105 -1.09	1.270 1.301 -2.38	1.167 1.176 -0.77	1.350 1.327 1.73	1.105 1.103 0.18	1.333 1.327 0.45	1.177 1.176 0.09	1.311 1.303 0.61	1.096 1.107 -0.99	1.278 1.261 1.35	0.990 0.967 2.38	
11		0.460 0.459 0.22	1.130 1.124 0.53	1.186 1.192 -0.50	1.047 1.077 -2.79	1.306 1.303 0.23	1.096 1.089 0.64	1.046 1.040 0.58	1.095 1.088 0.64	1.317 1.301 1.23	1.073 1.077 -0.37	1.188 1.196 -0.67	1.131 1.127 0.35	0.467 1.459 1.74	
12			0.410 0.418 -1.91	0.734 0.752 -2.39	1.166 1.196 -2.51	1.091 1.107 -1.45	1.077 1.071 0.56	1.053 1.047 0.57	1.084 1.071 1.21	1.114 1.105 0.81	1.198 1.192 0.50	0.744 0.752 -1.06	0.416 0.420 -0.95		
13				0.416 0.420 -0.95	1.131 1.127 0.35	1.264 1.261 0.24	1.127 1.114 1.17	1.306 1.285 1.63	1.131 1.113 1.62	1.278 1.259 1.51	1.124 1.124 0.00	0.413 0.418 -1.20			
14					0.460 0.459 0.22	0.971 0.967 0.41	1.074 1.056 1.70	0.873 0.863 1.16	1.078 1.057 1.99	0.982 0.967 1.55	0.465 0.459 1.31				
15							0.379 0.379 0.00	0.423 0.420 0.71	0.384 0.380 1.05						

INCORE
ANC
% DIFFERENCE

Mean Absolute Difference = 0.0100
Standard Deviation = 0.0127

BURNUP = 3260 MWD/MTU POWER LEVEL = 100.0% D BANK AT 230 STEPS

FIGURE 4.3-24

V.C. SUMMER NUCLEAR STATION CYCLE 5
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-SCM-05-015

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.396 0.402 -1.49	0.441 0.447 -1.34	0.401 0.401 0.00						
2					0.465 0.469 -0.85	0.957 0.969 -1.24	1.081 1.083 -0.18	0.878 0.884 -0.68	1.092 1.082 0.92	0.989 0.968 2.17	0.472 0.469 0.64				
3				0.434 0.434 0.00	1.123 1.120 0.27	1.266 1.268 -0.16	1.100 1.109 -0.81	1.284 1.287 -0.23	1.105 1.109 -0.36	1.270 1.269 0.08	1.123 1.122 0.09	0.434 0.436 -0.46			
4			0.435 0.436 -0.23	0.773 0.776 -0.39	1.216 1.218 -0.16	1.095 1.102 -0.64	1.052 1.059 -0.66	1.027 1.033 -0.58	1.047 1.058 -1.04	1.075 1.103 -2.54	1.212 1.221 -0.74	0.771 0.776 -0.64	0.432 0.434 -0.46		
5		0.417 0.469 0.43	1.130 1.122 0.71	1.232 1.221 0.90	1.089 1.084 0.46	1.313 1.302 0.84	1.060 1.069 -0.84	1.011 1.021 -0.98	1.049 1.069 -1.87	1.278 1.303 -1.92	1.066 1.084 -1.66	1.215 1.218 -0.25	1.117 1.120 -0.27	0.459 0.469 -2.13	
6		0.973 0.968 0.52	1.274 1.269 0.39	1.107 1.130 0.36	1.319 1.303 1.23	1.157 1.153 0.35	1.300 1.299 0.08	1.069 1.075 -0.56	1.284 1.299 -1.15	1.129 1.153 -2.08	1.292 1.302 -0.77	1.093 1.102 -0.82	1.266 1.263 -0.16	0.961 0.969 -0.83	
7	0.389 0.401 -2.99	1.083 1.082 0.09	1.118 1.109 0.81	1.071 1.058 1.23	1.087 1.069 1.68	1.318 1.299 1.46	1.109 1.114 -0.45	1.303 1.308 -0.38	1.096 1.114 -1.62	1.280 1.299 -1.46	1.051 1.069 -1.68	1.067 1.059 0.76	1.109 1.109 0.00	1.091 1.083 0.74	0.398 0.402 -1.00
8	0.434 0.447 -2.91	0.880 0.884 -0.45	1.304 1.287 1.32	1.037 1.033 0.39	1.019 1.021 -0.20	1.073 1.075 -0.19	1.313 1.308 0.38	1.020 1.027 -0.68	1.309 1.308 0.08	1.071 1.075 -0.37	1.020 1.021 -0.10	1.043 1.033 0.97	1.294 1.287 0.54	0.881 0.884 -0.34	0.442 0.447 -1.12
9	0.391 0.402 -2.74	1.090 1.083 0.65	1.127 1.109 1.62	1.066 1.059 0.66	1.066 1.069 -0.28	1.318 1.299 1.46	1.155 1.114 3.68	1.334 1.308 1.99	1.115 1.114 0.09	1.308 1.299 0.69	1.069 1.069 0.00	1.067 1.058 0.85	1.112 1.109 0.27	1.087 1.082 0.46	0.398 0.401 -0.75
10		0.986 0.969 1.75	1.290 1.268 1.74	1.098 1.102 -0.36	1.292 1.302 -0.77	1.159 1.153 0.52	1.329 1.299 2.31	1.085 1.075 0.93	1.319 1.299 1.54	1.162 1.153 0.78	1.315 1.303 0.92	1.087 1.103 -1.45	1.274 1.269 0.39	0.971 0.968 0.31	
11		0.469 0.469 0.00	1.125 1.120 0.45	1.217 1.218 -0.08	1.068 1.084 -1.48	1.317 1.303 1.07	1.079 1.069 0.94	1.029 1.021 0.78	1.081 1.069 1.12	1.313 1.302 0.84	1.075 1.084 -0.83	1.204 1.221 -1.39	1.115 1.122 -0.62	0.468 0.469 -0.21	
12			0.428 0.434 -1.38	0.765 0.776 -1.42	1.208 1.221 -1.06	1.093 1.103 -0.91	1.061 1.058 0.28	1.036 1.033 0.29	1.070 1.059 1.04	1.107 1.102 0.45	1.218 1.218 0.00	0.760 0.776 -2.06	0.427 0.436 -2.06		
13				0.429 0.436 -1.61	1.106 1.122 -1.43	1.260 1.269 -0.71	1.116 1.109 0.63	1.299 1.287 0.93	1.117 1.109 0.72	1.279 1.268 0.87	1.115 1.120 -0.45	0.425 0.434 -2.07			
14					0.461 0.469 -1.71	0.957 0.968 -1.14	1.087 1.082 0.46	0.884 0.884 0.00	1.097 1.083 1.29	0.977 0.969 0.83	0.473 0.469 0.85				
15							0.395 0.401 -1.50	0.444 0.447 -0.67	0.402 0.402 0.00						

Mean Absolute Difference = 0.0085
Stasndard Deviation = 0.0109

INCORE
ANC
% DIFFERENCE

BURNUP = 6174 MWD/MTU   POWER LEVEL = 100.0%   D BANK AT 230 STEPS

FIGURE 4.3-25

V.C. SUMMER NUCLEAR STATION CYCLE 5
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-05-018

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.414 0.419 -1.19	0.463 0.469 -1.28	0.416 0.417 -0.24						
2					0.468 0.477 -1.89	0.955 0.968 -1.34	1.097 1.097 0.00	0.897 0.901 -0.44	1.105 1.096 0.82	0.979 0.968 1.14	0.477 0.476 0.21				
3				0.442 0.446 -0.90	1.112 1.113 -0.09	1.260 1.262 -0.16	1.104 1.110 -0.54	1.289 1.288 0.08	1.109 1.110 -0.09	1.260 1.263 -0.24	1.113 1.114 -0.18	0.444 0.447 -0.67			
4			0.445 0.447 -0.45	0.786 0.789 -0.38	1.215 1.219 -0.33	1.092 1.099 -0.64	1.055 1.059 -0.38	1.031 1.036 -0.48	1.054 1.059 -0.47	1.079 1.100 -1.91	1.214 1.221 -0.57	0.784 0.789 -0.63	0.441 0.446 -1.12		
5		0.480 0.476 0.84	1.128 1.115 1.17	1.236 1.221 1.23	1.091 1.083 0.74	1.305 1.292 1.01	1.058 1.065 -0.66	1.014 1.020 -0.59	1.046 1.065 -1.78	1.268 1.293 -1.93	1.065 1.083 -1.66	1.217 1.219 -0.16	1.105 1.113 -0.72	0.463 0.477 -2.94	
6		0.975 0.968 0.72	1.271 1.263 0.63	1.107 1.100 0.[illegible]4	1.312 1.293 1.47	1.147 1.141 0.53	1.283 1.281 0.16	1.058 1.066 -0.75	1.261 1.281 -1.56	1.113 1.141 -2.45	1.277 1.292 -1.16	1.089 1.099 -0.91	1.256 1.262 -0.48	0.960 0.968 -0.83	
7	0.402 0.417 -3.60	1.095 1.096 -0.09	1.121 1.110 0.99	1.071 1.059 1.13	1.078 1.065 1.22	1.297 1.281 1.25	1.097 1.100 -0.27	1.281 1.286 -0.39	1.077 1.100 -2.09	1.257 1.281 -1.87	1.040 1.065 -2.35	1.059 1.059 0.00	1.107 1.110 -0.27	1.102 1.097 0.46	0.415 0.419 -0.95
8	0.451 0.469 -3.84	0.896 0.901 -0.55	1.308 1.288 1.55	1.039 1.036 0.29	1.015 1.020 -0.49	1.063 1.066 -0.28	1.295 1.286 0.70	1.008 1.016 -0.79	1.287 1.286 0.08	1.061 1.066 -0.47	1.017 1.020 -0.29	1.038 1.036 0.19	1.290 1.288 0.16	0.897 0.901 -0.44	0.463 0.469 -1.28
9	0.403 0.419 -3.82	1.102 1.097 0.46	1.129 1.110 1.71	1.066 1.059 0.66	1.060 1.065 -0.47	1.300 1.281 1.48	1.146 1.100 4.18	1.315 1.286 2.26	1.102 1.100 0.18	1.290 1.281 0.70	1.064 1.065 -0.09	1.061 1.059 0.19	1.107 1.110 -0.27	1.096 1.096 0.00	0.413 0.417 -0.96
10		0.984 0.968 1.65	1.282 1.262 1.58	1.096 1.099 -0.27	1.284 1.292 -0.62	1.15[illegible] 1.141 0.88	1.317 1.281 2.81	1.080 1.066 1.31	1.303 1.281 1.72	1.153 1.141 1.05	1.306 1.293 1.01	1.084 1.100 -1.45	1.261 1.263 -0.16	0.966 0.968 -0.21	
11		0.479 0.477 0.42	1.124 1.113 0.99	1.221 1.219 0.16	1.066 1.083 -1.57	1.312 1.293 1.47	1.083 1.065 1.69	1.036 1.020 1.57	1.081 1.065 1.50	1.308 1.292 1.24	1.075 1.083 -0.74	1.207 1.221 -1.15	1.109 1.115 -0.54	0.476 0.476 0.00	
12			0.445 0.446 -0.22	0.783 0.789 -0.76	1.207 1.221 -1.15	1.096 1.100 -0.36	1.068 1.059 0.85	1.[illegible]44 1.[illegible]36 0.77	1.071 1.059 1.13	1.104 1.099 0.45	1.218 1.219 -0.08	0.776 0.789 -1.65	0.440 0.447 -1.57		
13				0.440 0.447 -1.57	1.090 1.115 -2.24	1.254 1.263 -0.71	1.124 1.110 1.26	1.307 1.288 1.48	1.119 1.110 0.81	1.271 1.262 0.71	1.102 1.113 -0.99	0.438 0.446 -1.79			
14					0.463 0.476 -2.73	0.955 0.968 -1.34	1.104 1.096 0.73	0.906 0.901 0.55	1.111 1.097 1.28	0.976 0.968 0.83	0.476 0.477 -0.21				
15							0.414 0.417 -0.72	0.469 0.469 0.00	0.423 0.419 0.95						

Mean Absolute Difference = 0.0092
Standard Deviation = 0.0120

INCORE
ANC
% DIFFERENCE

BURNUP = 8168 MWD/MTU   POWER LEVEL = 99.8%   D BANK AT 230 STEPS

FIGURE 4.3-26

V.C. SUMMER NUCLEAR STATION CYCLE 5
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-05-020

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.433 0.438 -1.14	0.487 0.493 -1.22	0.436 0.436 0.00						
2					0.486 0.486 0.00	0.963 0.968 -0.52	1.112 1.108 0.36	0.919 0.919 0.00	1.118 0.106 1.08	0.983 0.967 1.65	0.489 0.485 0.82				
3				0.461 0.458 0.66	1.119 1.107 1.08	1.260 1.253 0.56	1.112 1.112 0.00	1.292 1.285 0.54	1.113 1.112 0.09	1.254 1.253 0.08	1.112 1.108 0.36	0.460 0.460 0.00			
4			0.462 0.460 0.43	0.803 0.803 0.00	1.219 1.215 0.33	0.090 1.096 -0.55	1.059 1.061 -0.19	1.038 1.040 -0.19	1.058 1.061 -0.28	1.076 1.097 -1.91	1.213 1.217 -0.33	0.800 0.803 -0.37	0.457 0.458 -0.22		
5		0.489 0.485 0.82	1.121 1.108 1.17	1.231 1.217 1.15	1.088 1.082 0.55	1.287 1.279 0.63	1.050 1.062 -1.13	1.010 1.022 -1.17	1.044 1.063 -1.79	1.257 1.279 -1.72	1.066 1.082 -1.48	1.214 1.215 -0.08	1.106 1.107 -0.09	0.476 0.486 -2.06	
6		[illegible] 0.967 0.41	1.255 1.253 0.16	1.099 1.097 0.18	1.293 1.279 1.09	1.131 1.130 0.09	1.257 1.262 -0.40	1.048 1.059 -1.04	1.244 1.262 -1.43	1.108 1.130 -1.95	1.267 1.279 -0.94	1.087 1.096 -0.82	1.252 1.253 -0.08	0.965 0.968 -0.31	
7	0.430 0.436 -1.38	1.108 1.106 0.18	1.116 1.112 0.36	1.071 1.061 0.94	1.078 1.063 1.41	1.279 1.262 1.25	1.080 1.089 -0.83	1.257 1.264 -0.55	1.069 1.089 -1.84	1.243 1.262 -1.51	1.041 1.062 -1.98	1.047 1.061 -1.32	1.102 1.112 -0.90	1.107 1.108 -0.09	0.435 0.438 -0.68
8	0.486 0.493 -1.42	0.918 0.919 -0.11	1.297 1.285 0.93	1.039 1.040 -0.10	1.015 1.022 -0.68	1.054 1.059 -0.47	1.268 1.264 0.32	1.000 1.008 -0.79	1.262 1.264 -0.16	1.053 1.059 -0.57	1.017 1.022 -0.49	1.025 1.040 -1.44	1.279 1.285 -0.47	0.914 0.919 -0.54	0.491 0.493 -0.41
9	0.431 0.438 -1.60	1.119 1.108 0.99	1.130 1.112 1.62	1.066 1.061 0.47	1.054 1.062 -0.75	1.277 1.262 1.19	1.133 1.089 4.04	1.287 1.264 1.82	1.085 1.089 -0.37	1.265 1.262 0.24	1.057 1.063 -0.56	1.042 1.061 -1.79	1.104 1.112 -0.72	1.109 1.106 0.27	0.436 0.436 0.00
10		0.985 0.968 1.76	1.274 1.253 1.68	1.089 1.096 -0.64	1.262 1.279 -1.33	1.134 1.130 0.35	1.295 1.262 2.61	1.070 1.059 1.04	1.279 1.262 1.35	1.135 1.130 0.44	1.285 1.279 0.47	1.075 1.097 -2.01	1.257 1.253 0.32	0.979 0.967 1.24	
11		0.488 0.486 0.41	1.118 1.107 0.99	1.217 1.215 0.16	1.061 1.082 -1.94	1.290 1.279 0.86	1.075 1.063 1.13	1.033 1.022 1.08	1.076 1.062 1.32	1.288 1.279 0.70	1.069 1.082 -1.20	1.201 1.217 -1.31	1.105 1.108 -0.27	0.490 0.485 1.03	
12			0.457 0.458 -0.22	0.795 0.803 -1.00	1.200 1.217 -1.40	1.084 1.097 -1.19	1.066 1.061 0.47	1.045 1.040 0.48	1.073 1.061 1.13	1.097 1.096 0.09	1.211 1.215 -0.33	0.788 0.803 -1.87	0.454 0.460 -1.30		
13				0.458 0.460 -0.43	1.105 1.108 -0.27	1.250 1.253 -0.24	1.120 1.112 0.72	1.301 1.285 1.25	1.122 1.112 0.90	1.264 1.253 0.88	1.096 1.107 -0.99	0.451 0.458 -1.53			
14					0.483 0.485 -0.41	0.967 0.967 0.00	1.120 1.106 1.27	0.927 0.919 0.87	1.124 1.108 1.44	0.977 0.968 0.93	0.486 0.486 0.00				
15							0.435 0.436 -0.23	0.495 0.493 0.41	0.442 0.438 0.91						

Mean Absolute Difference = 0.0084
Standard Deviation = 0.0110

INCORE ANC % DIFFERENCE

BURNUP = 10539 MWD/MTU   POWER LEVEL = 99.9%   D BANK AT 230 STEPS

FIGURE 4.3-27

V.C. SUMMER NUCLEAR STATION CYCLE 5
RADIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC FOR MAP-FCFM-05-022

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.455 0.456 -0.22	0.516 0.517 -0.19	0.458 0.455 0.66						
2					0.500 0.495 1.01	0.957 0.968 -1.14	1.121 1.117 0.36	0.937 0.937 0.00	1.132 1.115 1.52	0.988 0.968 2.07	0.499 0.495 0.81				
3				0.476 0.471 1.06	1.119 1.102 1.54	1.244 1.242 0.16	1.111 1.114 -0.27	1.285 1.282 0.23	1.115 1.114 0.09	1.242 1.242 0.00	1.105 1.103 0.18	0.472 0.472 0.00			
4			0.477 0.472 1.06	0.819 0.815 0.49	1.216 1.208 0.66	1.086 1.093 -0.64	1.059 1.065 -0.56	1.041 1.046 -0.48	1.059 1.064 -0.47	1.067 1.094 -2.47	1.202 1.210 -0.66	0.811 0.815 -0.49	0.471 0.471 0.00		
5		0.500 0.495 1.01	1.120 1.103 1.54	1.224 1.210 1.16	1.086 1.079 0.65	1.271 1.263 0.63	1.049 1.061 -1.13	1.014 1.026 -1.17	1.038 1.062 -2.26	1.231 1.263 -2.53	1.058 1.079 -1.95	1.206 1.208 -0.17	1.106 1.102 0.36	0.489 0.495 -1.21	
6		0.973 0.968 0.52	1.248 1.242 0.48	1.098 1.094 0.37	1.276 1.263 1.03	1.120 1.121 -0.09	1.237 1.244 -0.56	1.043 1.055 -1.14	1.222 1.244 -1.77	1.062 1.121 -2.59	1.244 1.263 -1.50	1.082 1.093 -1.01	1.244 1.242 0.16	0.972 0.968 0.41	
7	0.445 0.455 -2.20	1.115 1.115 0.00	1.118 1.114 0.36	1.072 1.064 0.75	1.072 1.062 0.94	1.255 1.244 0.88	1.070 1.080 -0.93	1.236 1.243 -0.56	1.056 1.080 -2.22	1.218 1.244 -2.09	1.032 1.061 -2.73	1.056 1.065 -0.85	1.110 1.114 -0.36	1.123 1.117 0.54	0.456 0.456 0.00
8	0.507 0.517 -1.93	0.933 0.937 -0.43	1.294 1.282 0.94	1.043 1.046 -0.29	1.017 1.026 -0.88	1.047 1.055 -0.76	1.243 1.243 0.00	0.991 1.003 -1.20	1.236 1.243 -0.56	1.043 1.055 -1.14	1.015 1.026 -1.07	1.038 1.046 -0.76	1.282 1.282 0.00	0.938 0.937 0.11	0.518 0.517 0.19
9	0.446 0.456 -2.19	1.126 1.117 0.81	1.131 1.114 1.53	1.068 1.065 0.28	1.051 1.061 -0.94	1.255 1.244 0.88	1.117 1.080 3.43	1.259 1.243 1.29	1.069 1.080 -1.02	1.239 1.244 -0.40	1.052 1.062 -0.94	1.058 1.064 -0.56	1.112 1.114 -0.18	1.122 1.115 0.63	0.456 0.455 0.22
10		0.987 0.968 1.96	1.265 1.242 1.85	1.008 1.093 -0.46	1.248 1.263 -1.19	1.124 1.121 0.27	1.271 1.244 2.17	1.060 1.055 0.47	1.252 1.244 0.64	1.120 1.121 -0.09	1.266 1.263 0.24	1.075 1.094 -1.74	1.248 1.242 0.48	0.983 0.968 1.55	
11		0.502 0.495 1.41	1.122 1.102 1.81	1.217 1.208 0.75	1.061 1.079 -1.67	1.276 1.263 1.03	1.073 1.062 1.04	1.036 1.026 0.97	1.068 1.061 0.66	1.267 1.263 0.32	1.066 1.079 -1.20	1.199 1.210 -0.91	1.105 1.103 0.18	0.502 0.495 1.41	
12			0.477 0.417 1.27	0.814 0.815 -0.12	1.195 1.210 -1.24	1.087 1.094 -0.64	1.068 1.064 0.38	1.049 1.046 0.29	1.071 1.065 0.56	1.092 1.093 -0.09	1.204 1.208 -0.33	0.805 0.815 -1.23	0.470 0.472 -0.42		
13				0.474 0.472 0.42	1.104 1.103 0.09	1.245 1.242 0.24	1.123 1.114 0.81	1.299 1.282 1.33	1.124 1.114 0.90	1.255 1.242 1.05	1.094 1.102 -0.73	0.467 0.471 -0.85			
14					0.492 0.495 -0.61	0.971 0.968 0.31	1.128 1.115 1.17	0.946 0.937 0.96	1.133 1.117 1.43	0.980 0.968 1.24	0.495 0.495 0.00				
15							0.456 0.455 0.22	0.521 0.517 0.77	0.463 0.456 1.54						

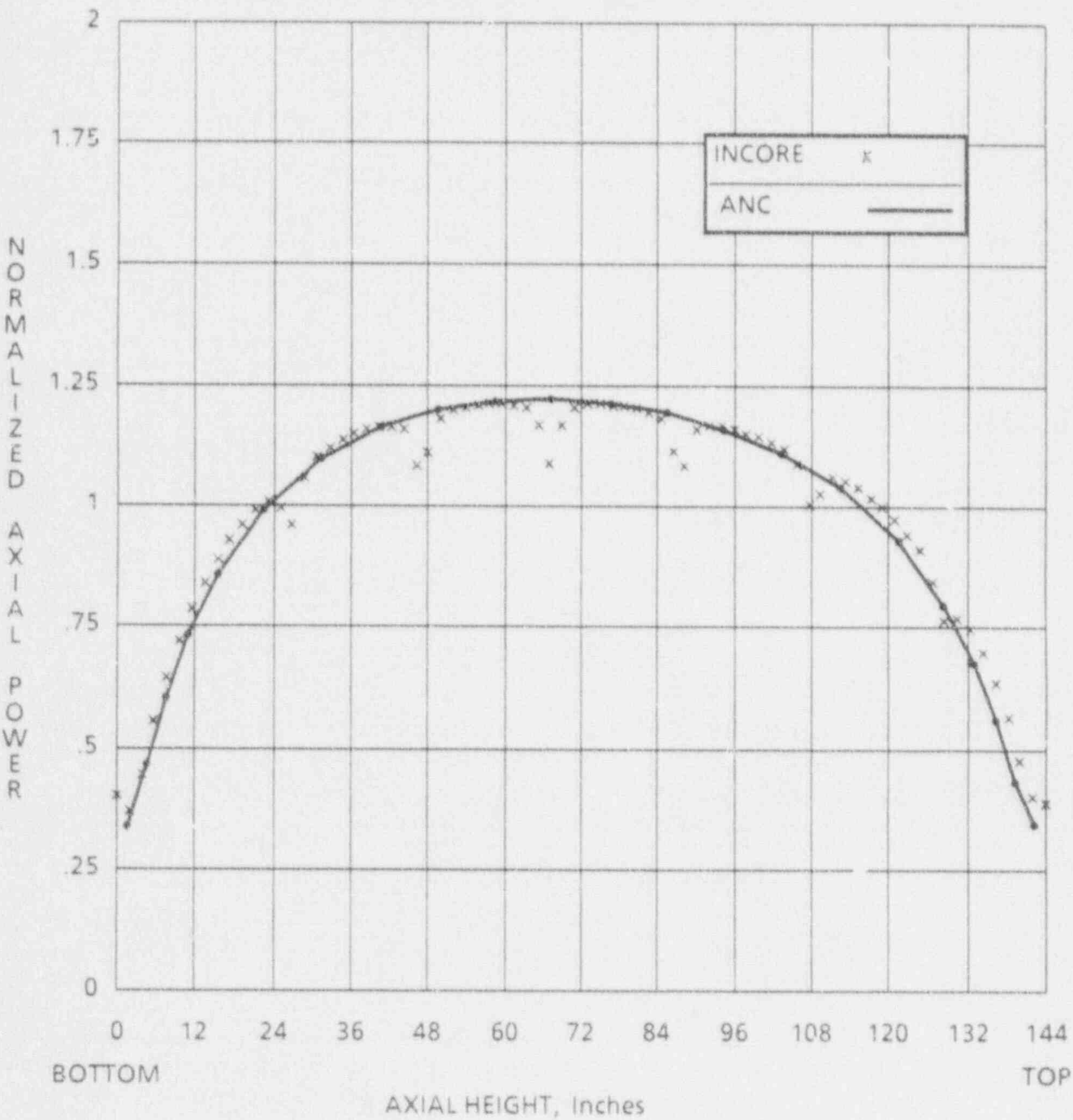
Mean Absolute Difference = 0.0087
Standard Deviation = 0.0114

INCORE
ANC
% DIFFERENCE

BURNUP = 12793 MWD/MTU   POWER LEVEL = 100.1%   D BANK AT 230 STEPS

FIGURE 4.3-28

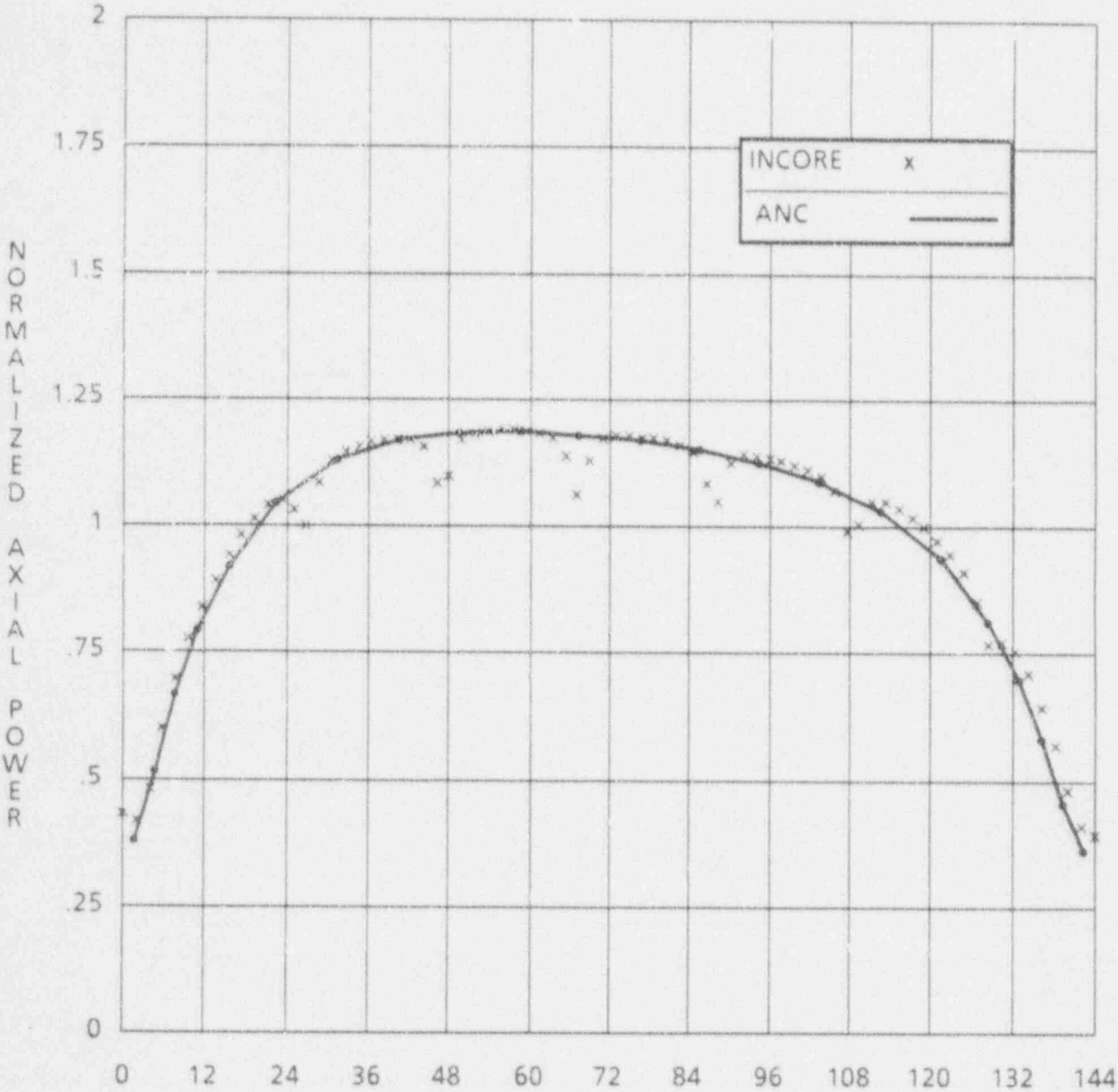
V.C. SUMMER NUCLEAR STATION CYCLE 3
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-03-007



BURNUP = 1839 MWD/MTU POWER LEVEL = 100.0% D BANK AT 228 STEPS

FIGURE 4.3-29

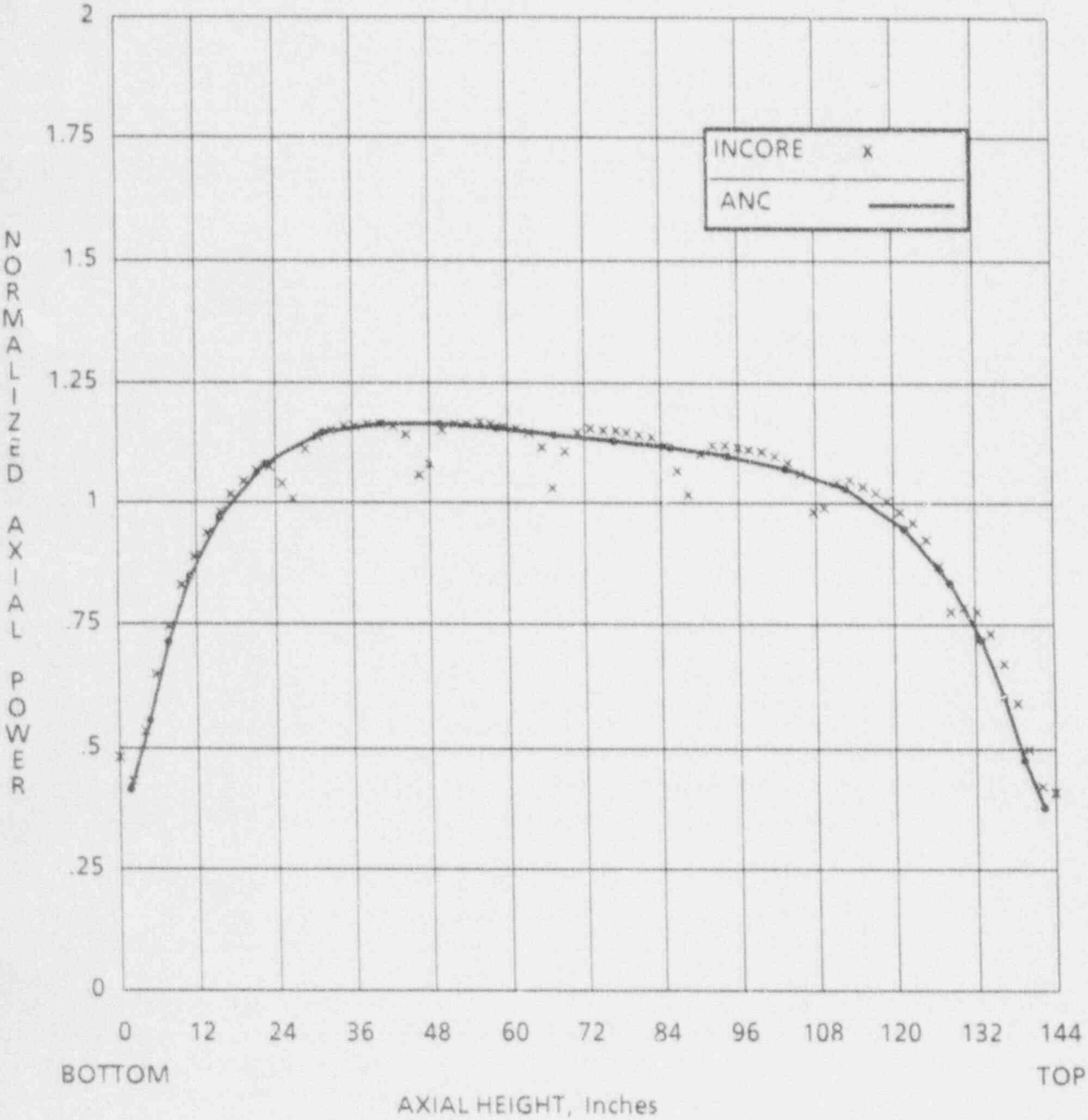
V.C. SUMMER NUCLEAR STATION CYCLE 3
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-03-014



BURNUP = 3920 MWD/MTU POWER LEVEL = 99.0% D BANK AT 228 STEPS

FIGURE 4.3-30

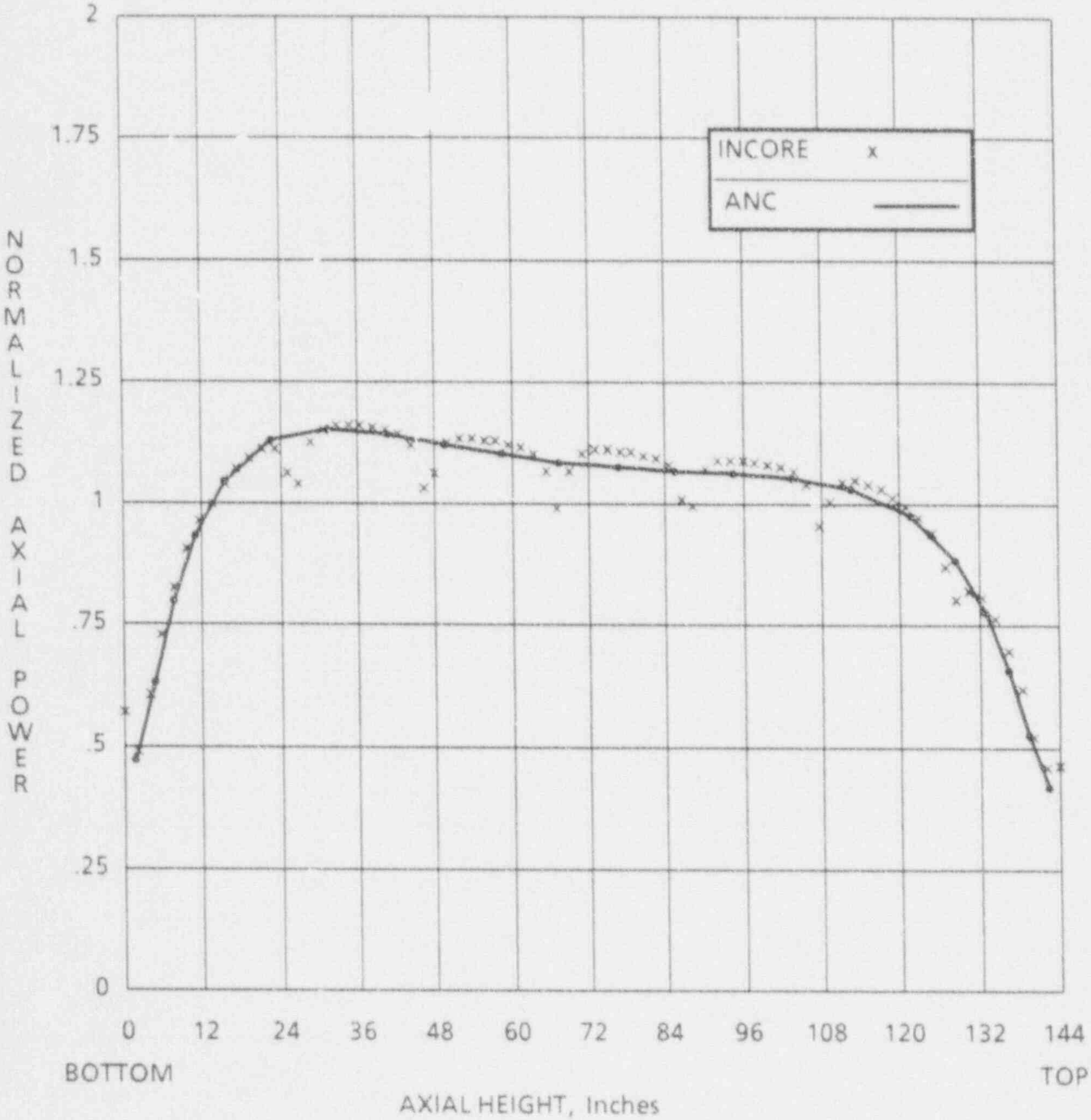
V.C. SUMMER NUCLEAR STATION CYCLE 3
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-03-017



BURNUP = 5558 MWD/MTU POWER LEVEL = 99.8% D BANK AT 228 STEPS

FIGURE 4.3-31

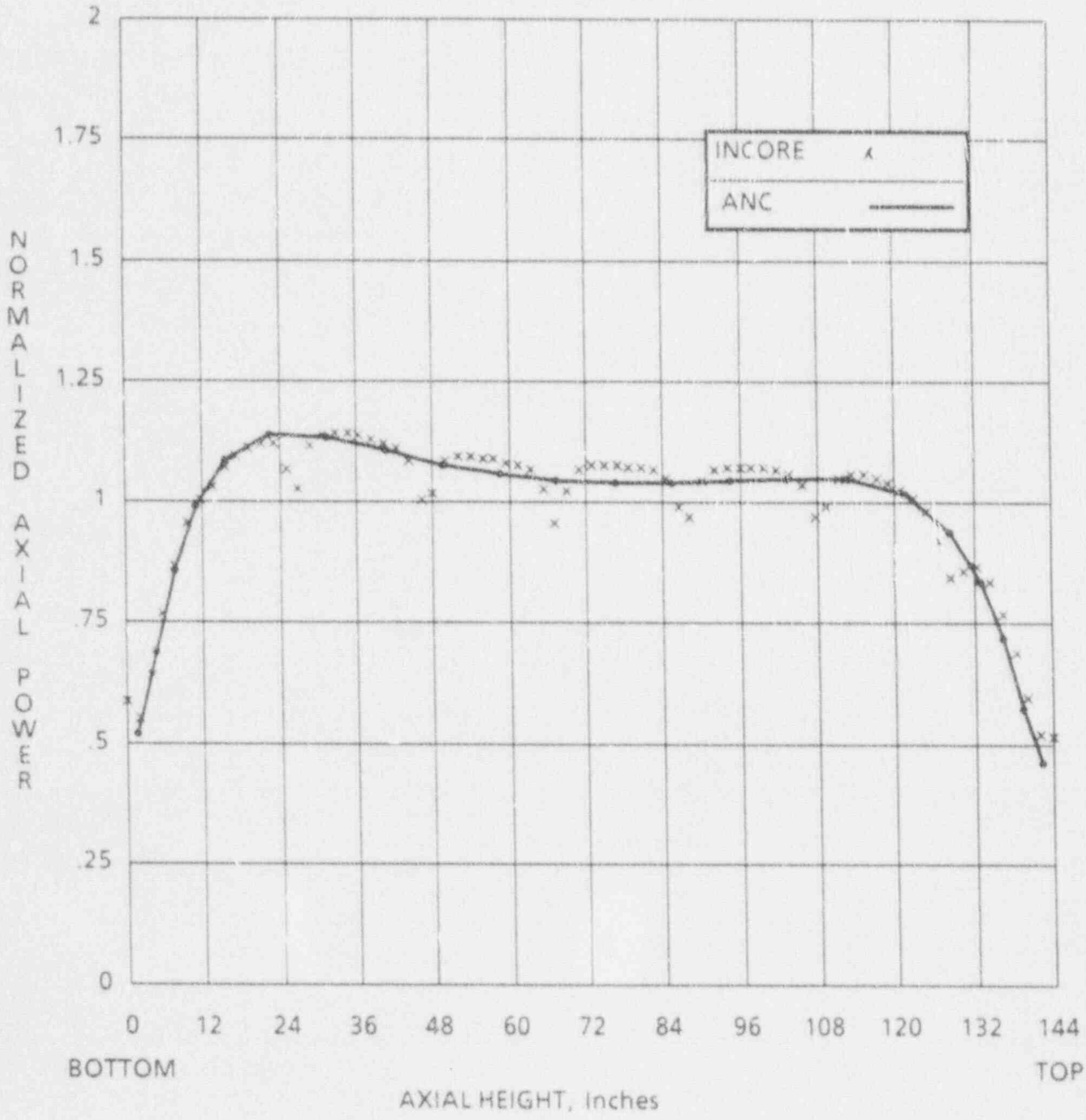
V.C. SUMMER NUCLEAR STATION CYCLE 3
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-03-025



BURNUP = 8689 MWD/MTU POWER LEVEL = 100.0% D BANK AT 228 STEPS

FIGURE 4.3-32

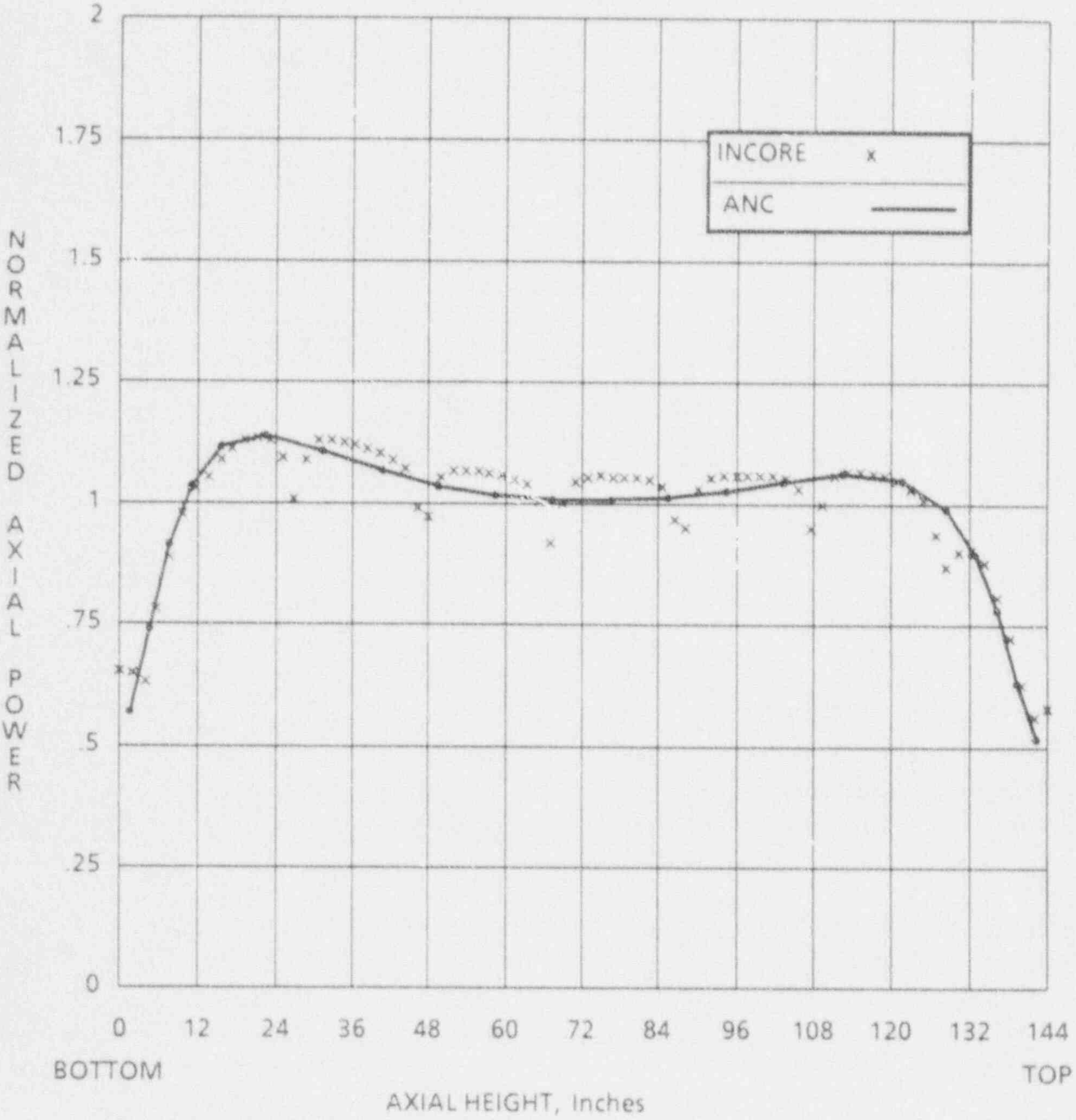
V.C. SUMMER NUCLEAR STATION CYCLE 3
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-03-031



BURNUP = 11123 MWD/MTU POWER LEVEL = 99.9% D BANK AT 228 STEPS

F'GURE 4.3-33

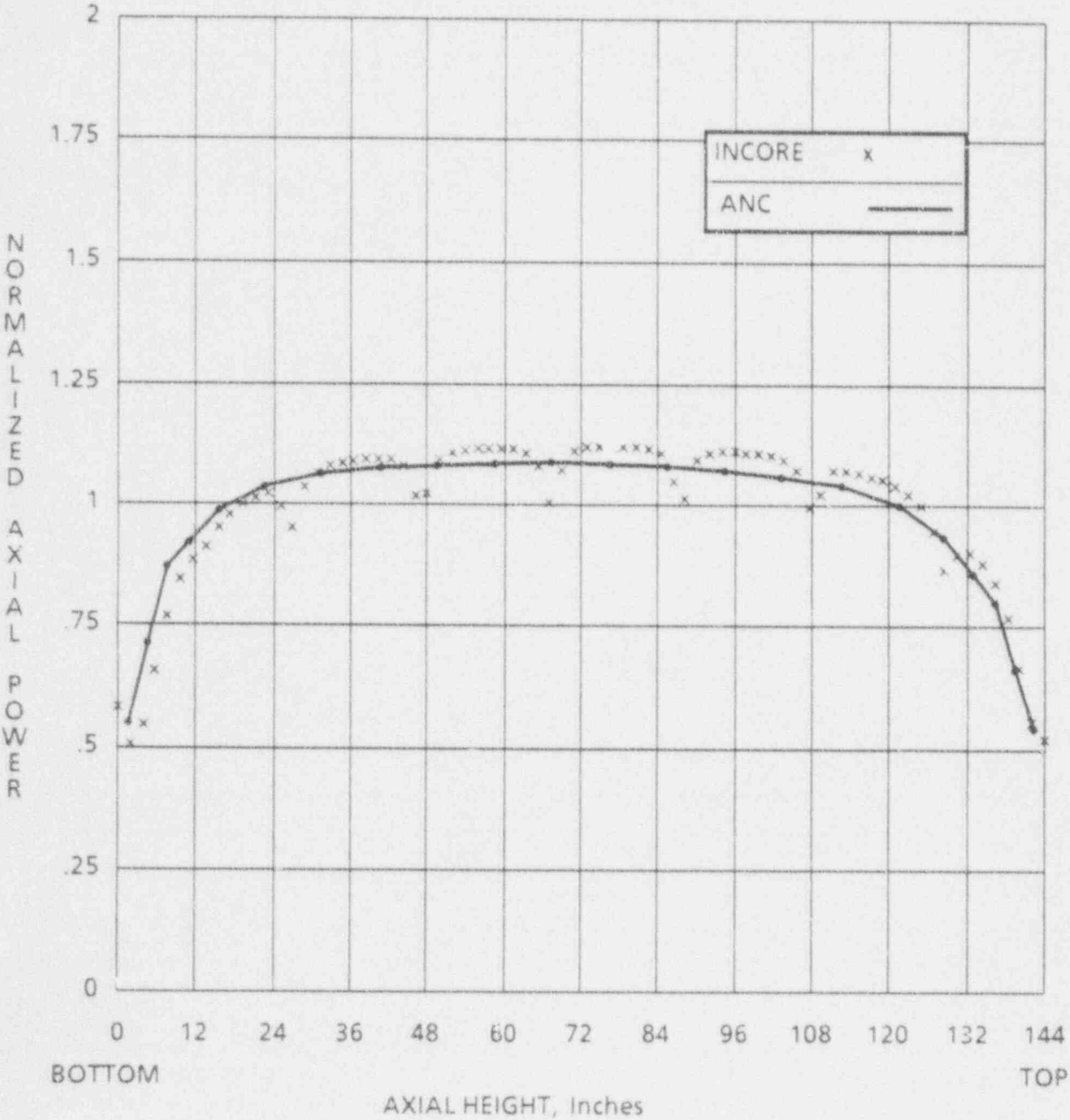
V.C. SUMMER NUCLEAR STATION CYCLE 3
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-03-037



BURNUP = 13581 MWD/MTU POWER LEVEL = 100.0% D BANK AT 228 STEPS

FIGURE 4.3-34

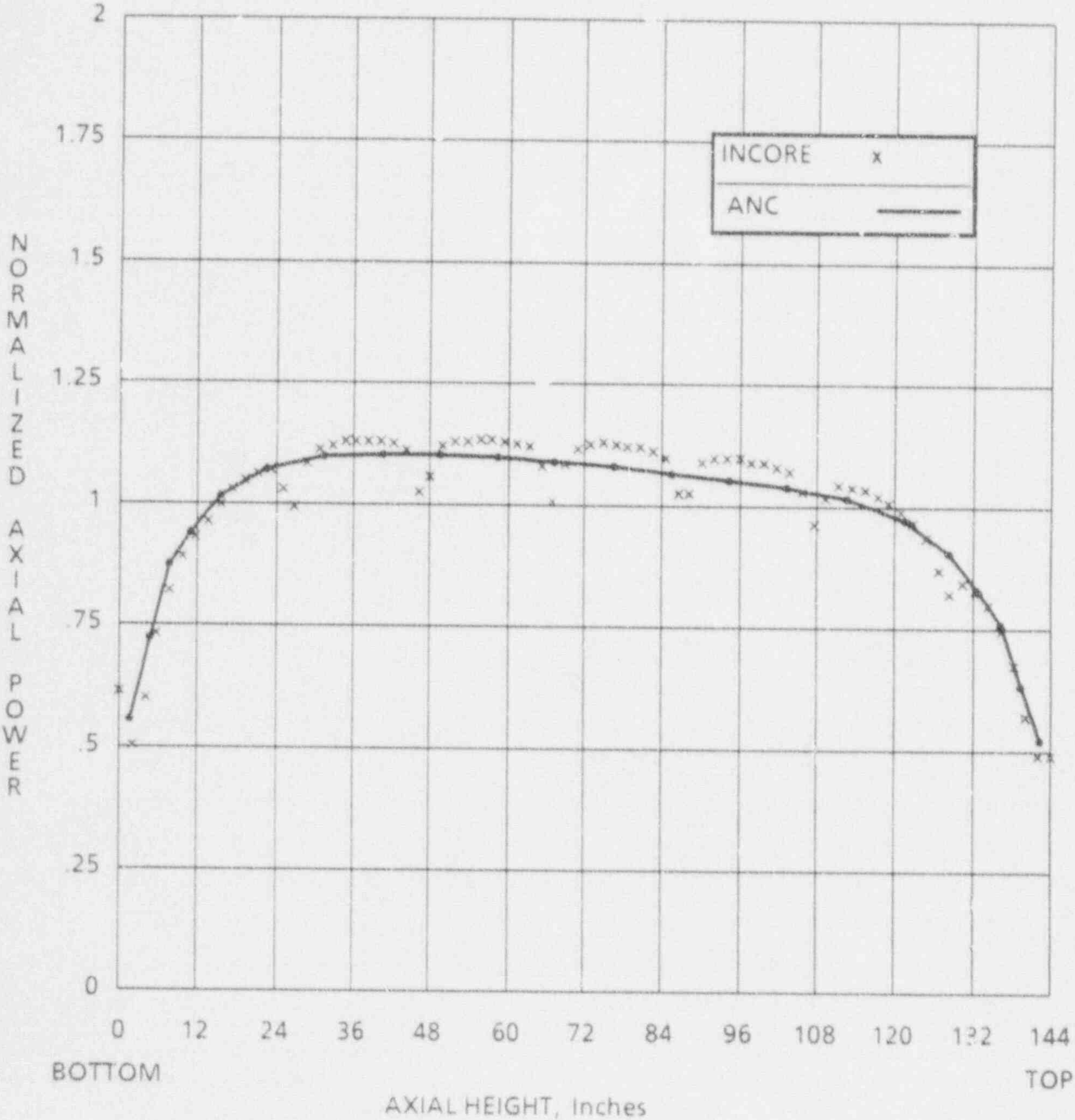
V.C. SUMMER NUCLEAR STATION CYCLE 4
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-04-005



BURNUP = 1232 MWD/MTU POWER LEVEL = 99.8% D BANK AT 228 STEPS

FIGURE 4.3-35

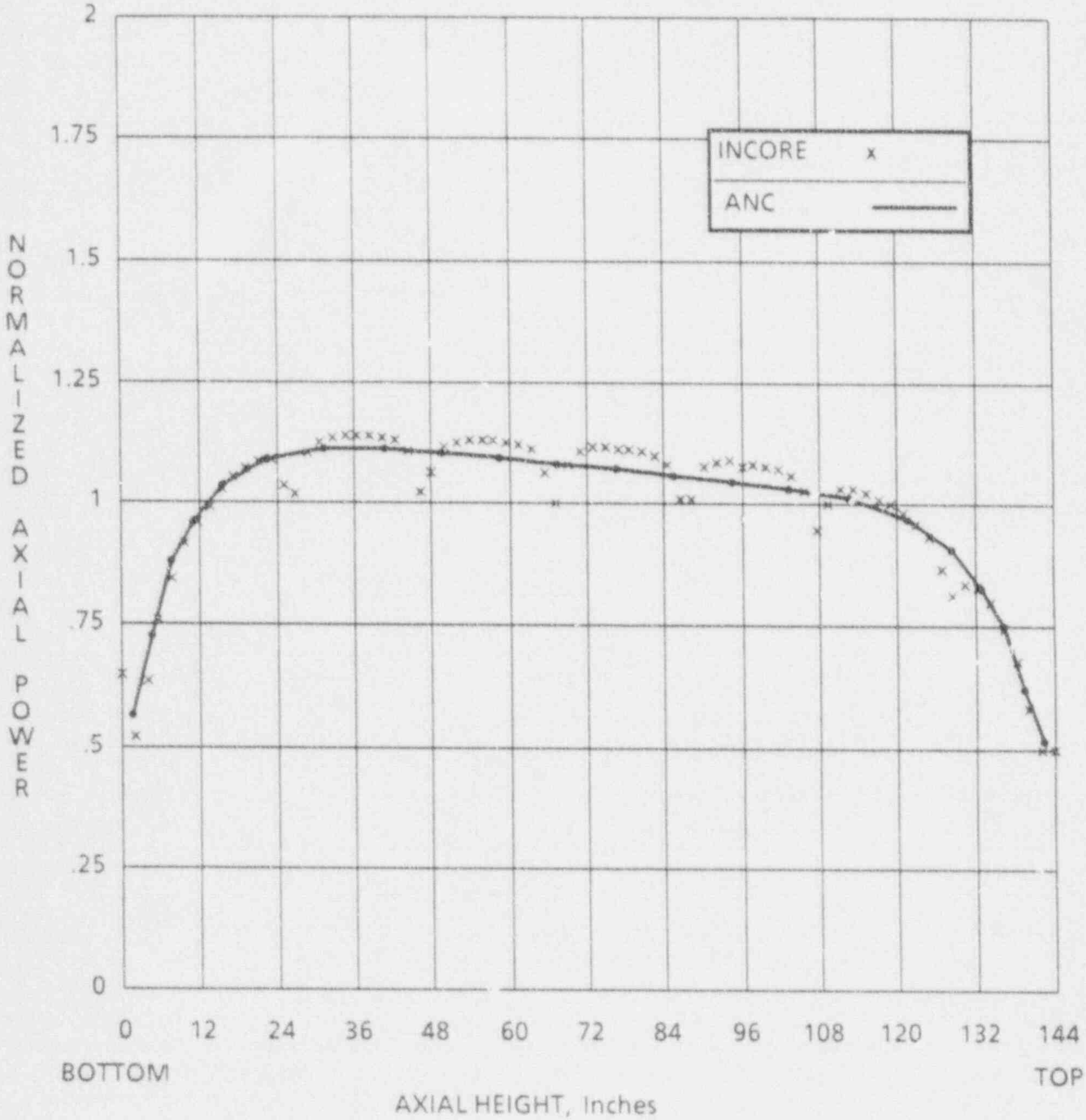
V.C. SUMMER NUCLEAR STATION CYCLE 4
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-04-010



BURNUP = 4154 MWD/MTU POWER LEVEL = 100.0% D BANK AT 228 STEPS

FIGURE 4.3-36

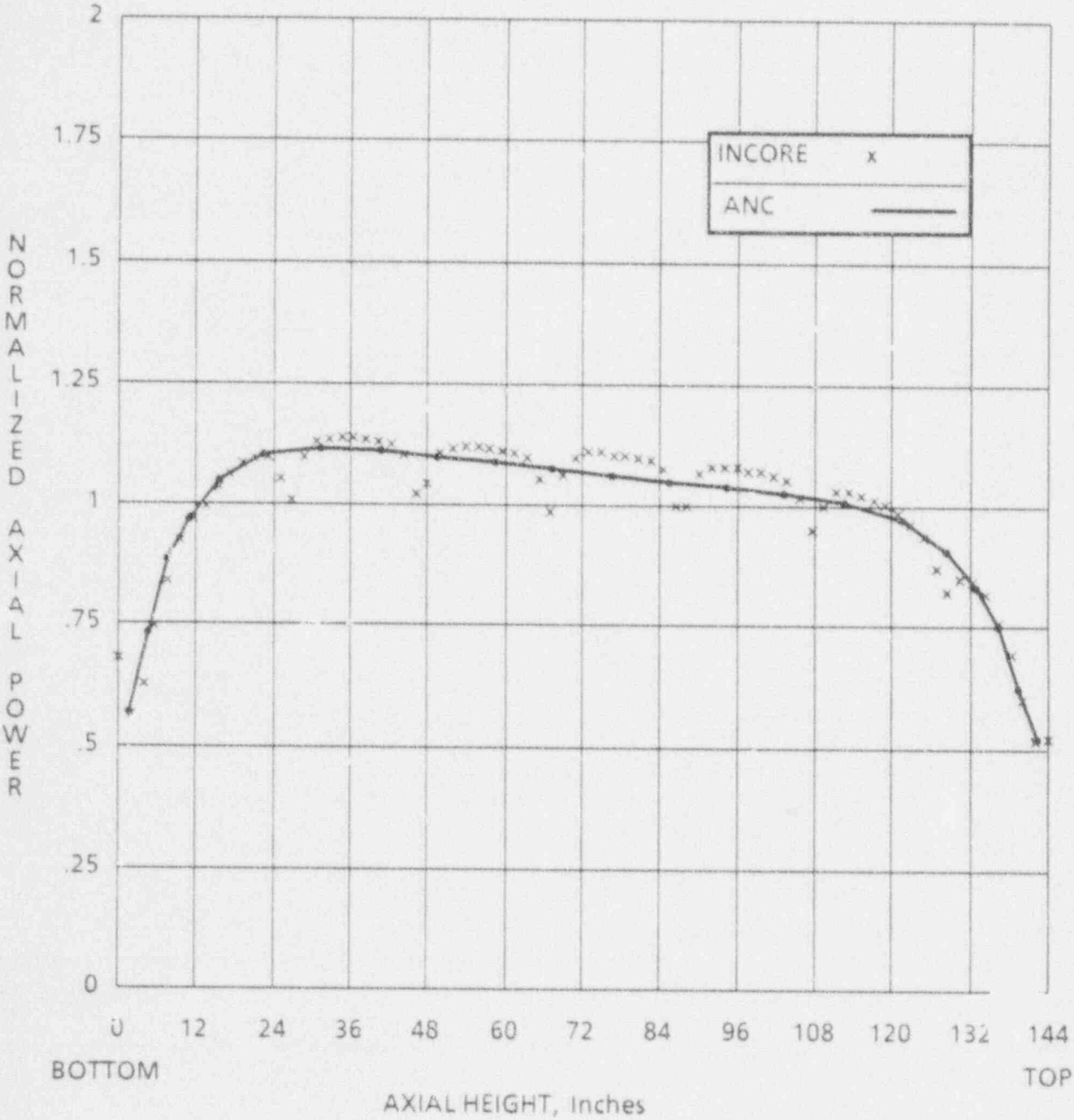
V.C. SUMMER NUCLEAR STATION CYCLE 4
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-04-012



BURNUP = 6398 MWD/MTU POWER LEVEL = 99.8% D BANK AT 228 STEPS

FIGURE 4.3-37

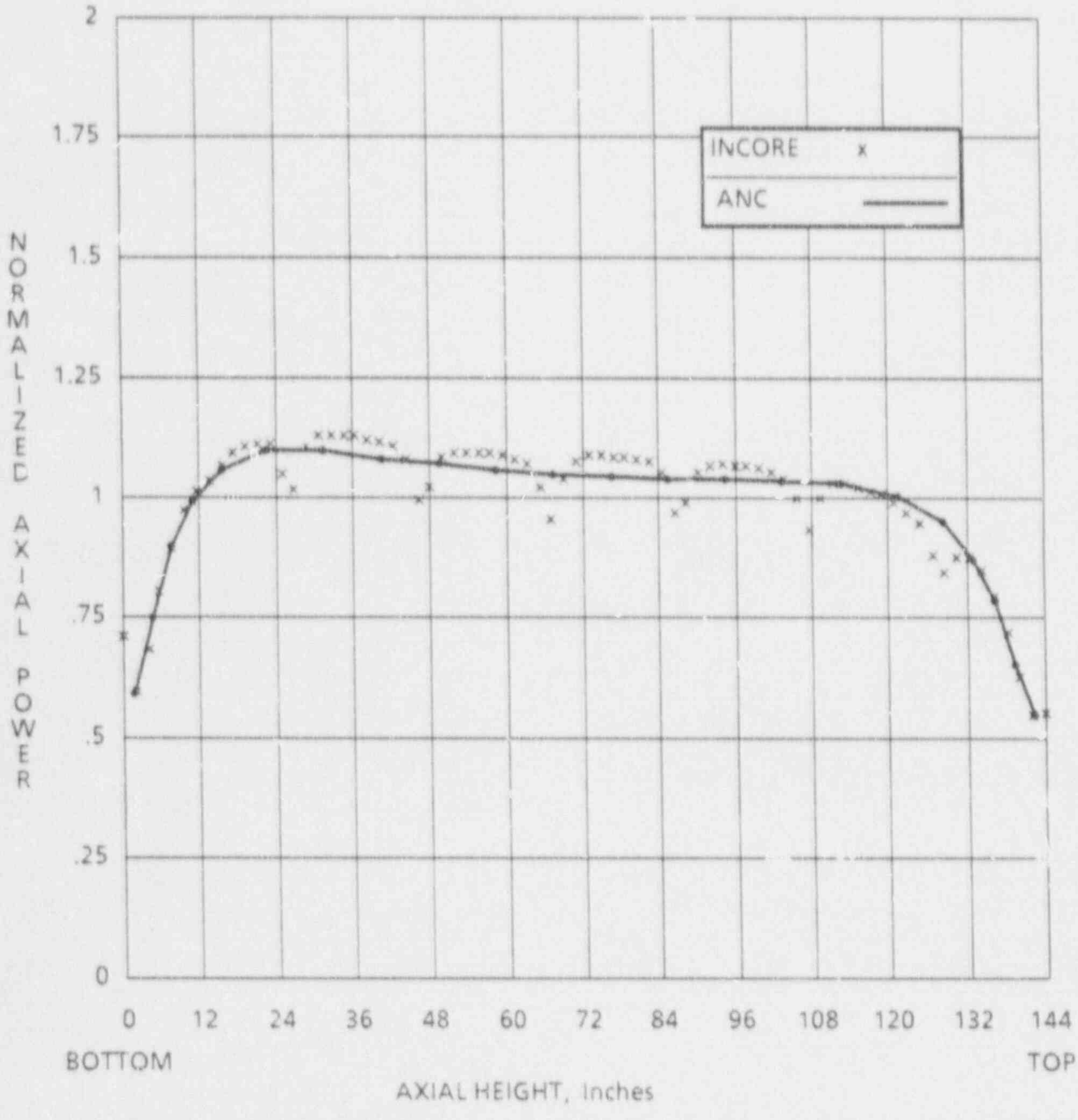
V.C. SUMMER NUCLEAR STATION CYCLE 4
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-04-016



BURNUP = 8608 MWD/MTU POWER LEVEL = 100.1% D BANK AT 228 STEPS

FIGURE 4.3-38

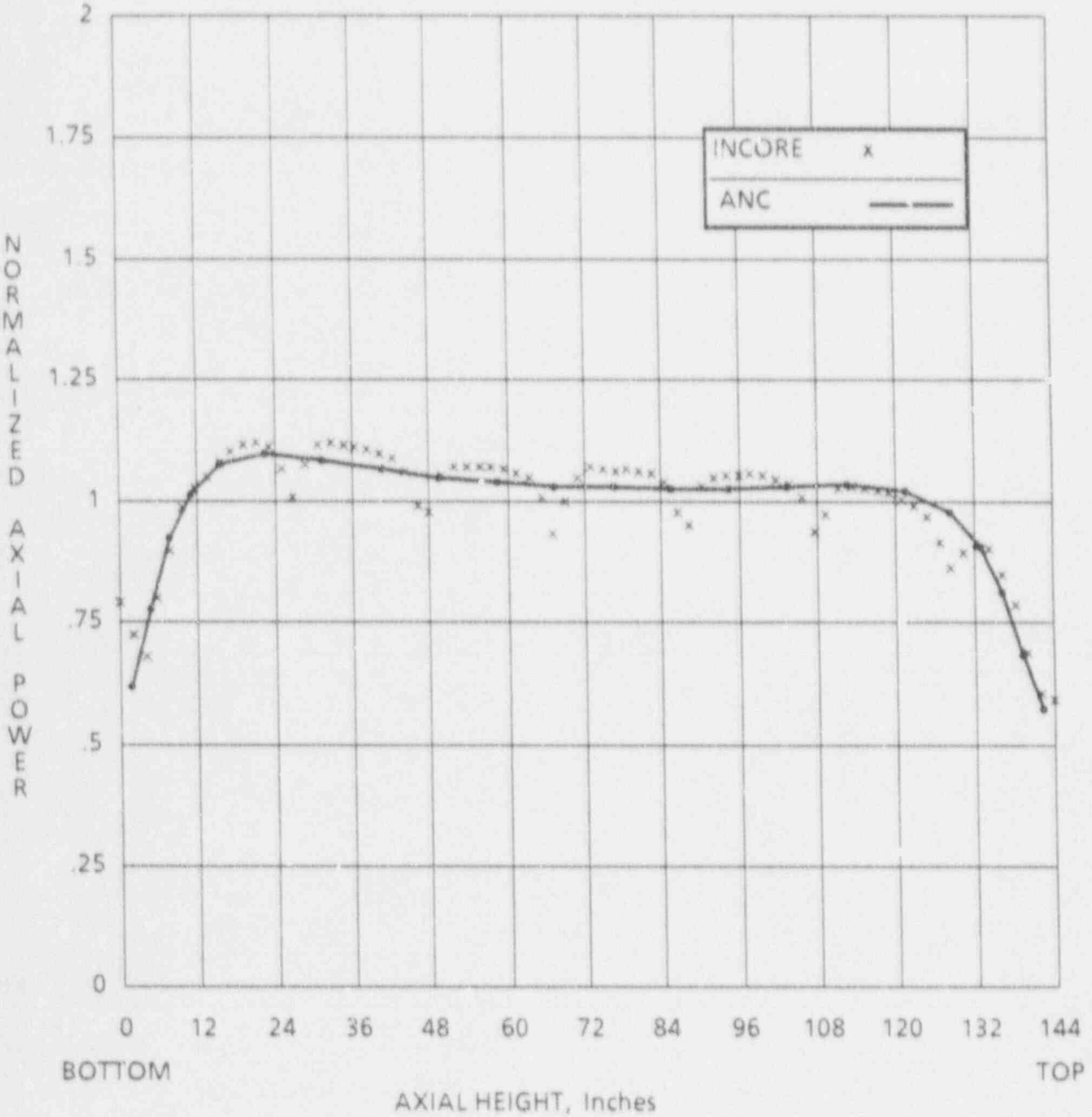
V.C. SUMMER NUCLEAR STATION CYCLE 4
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-04-021



BURNUP = 12192 MWD/MTU POWER LEVEL = 99.9% D BANK AT 226 STEPS

FIGURE 4.3-39

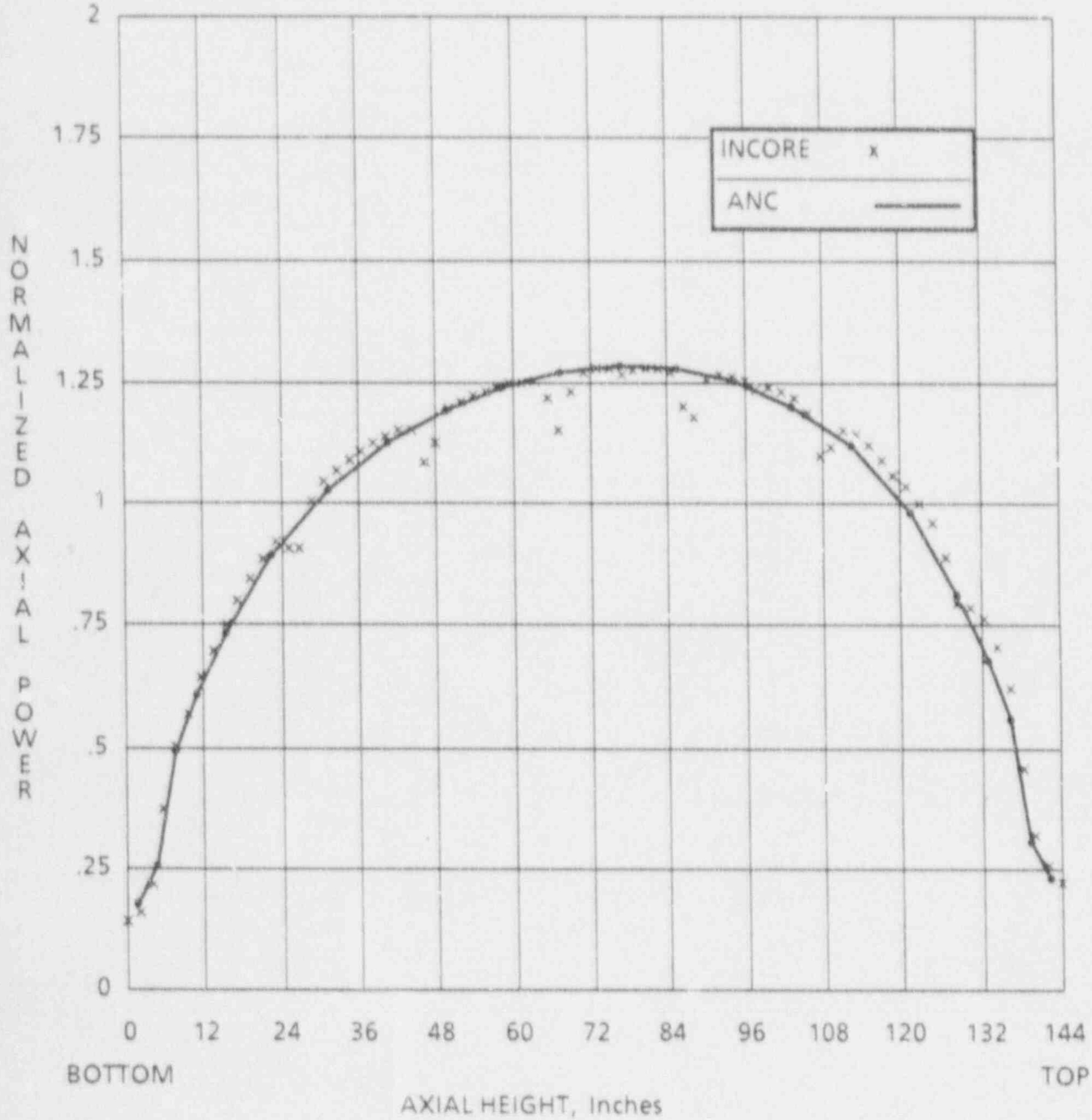
V.C. SUMMER NUCLEAR STATION CYCLE 4
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-04-026



BURNUP = 14970 MWD/MTU POWER LEVEL = 100.0% D BANK AT 228 STEPS

FIGURE 4.3-40

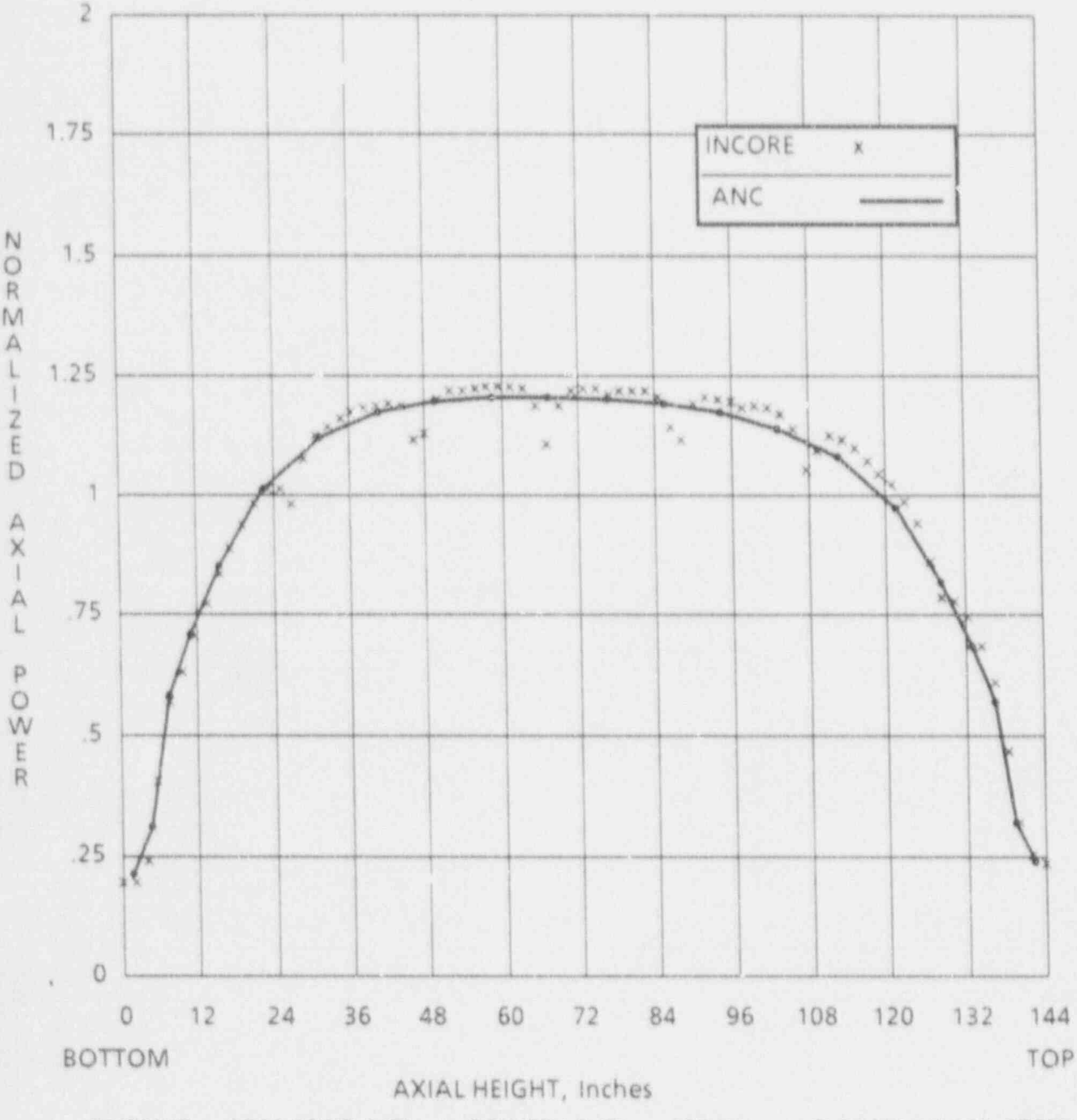
V.C. SUMMER NUCLEAR STATION CYCLE 5
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-05-006



BURNUP = 425 MWD/MTU POWER LEVEL = 99.9% D BANK AT 230 STEPS

FIGURE 4.3-41

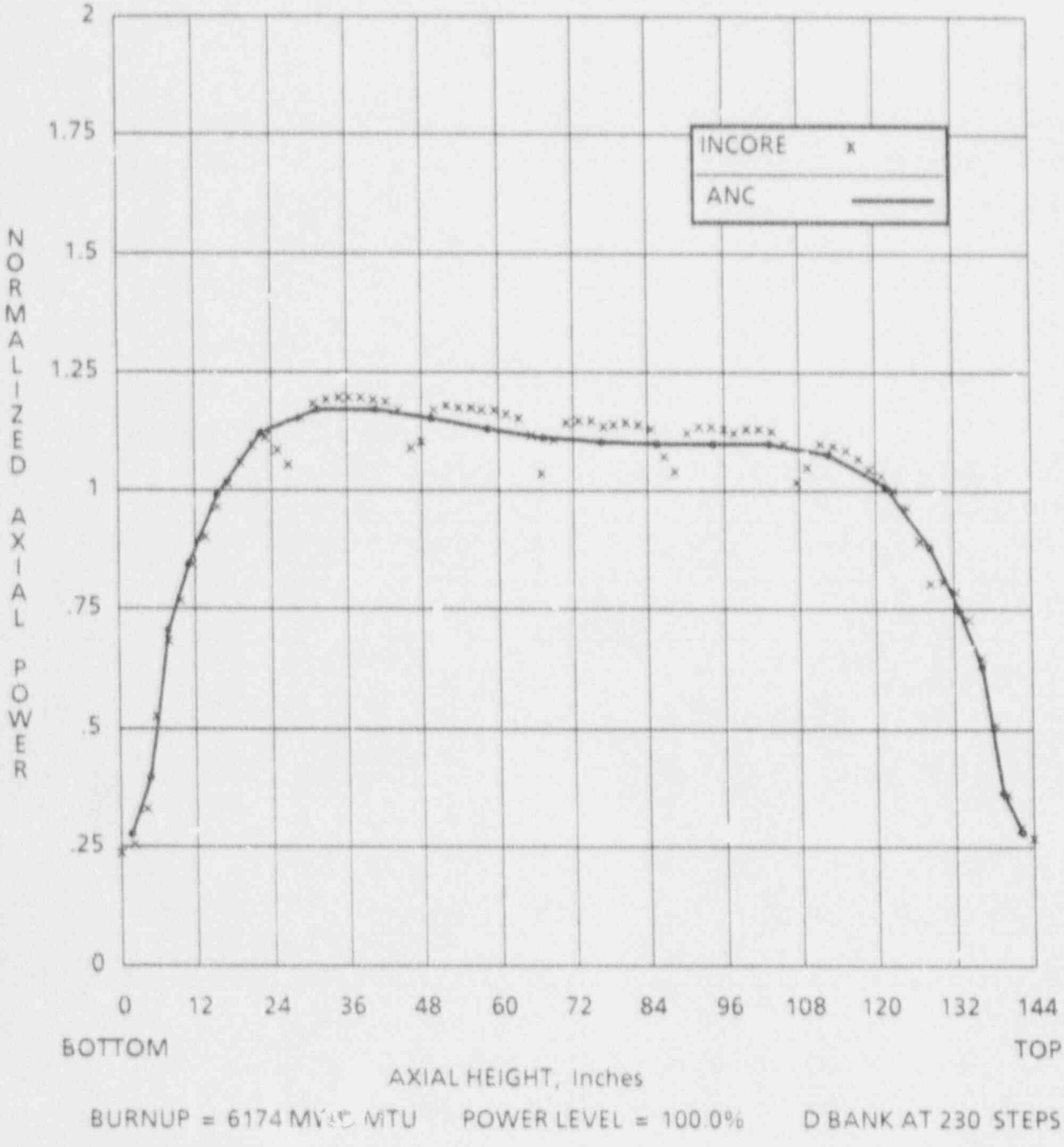
V.C. SUMMER NUCLEAR STATION CYCLE 5
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-05-012



BURNUP = 3260 MWD/MTU POWER LEVEL = 100.0% D BANK AT 230 STEPS

FIGURE 4.3-42

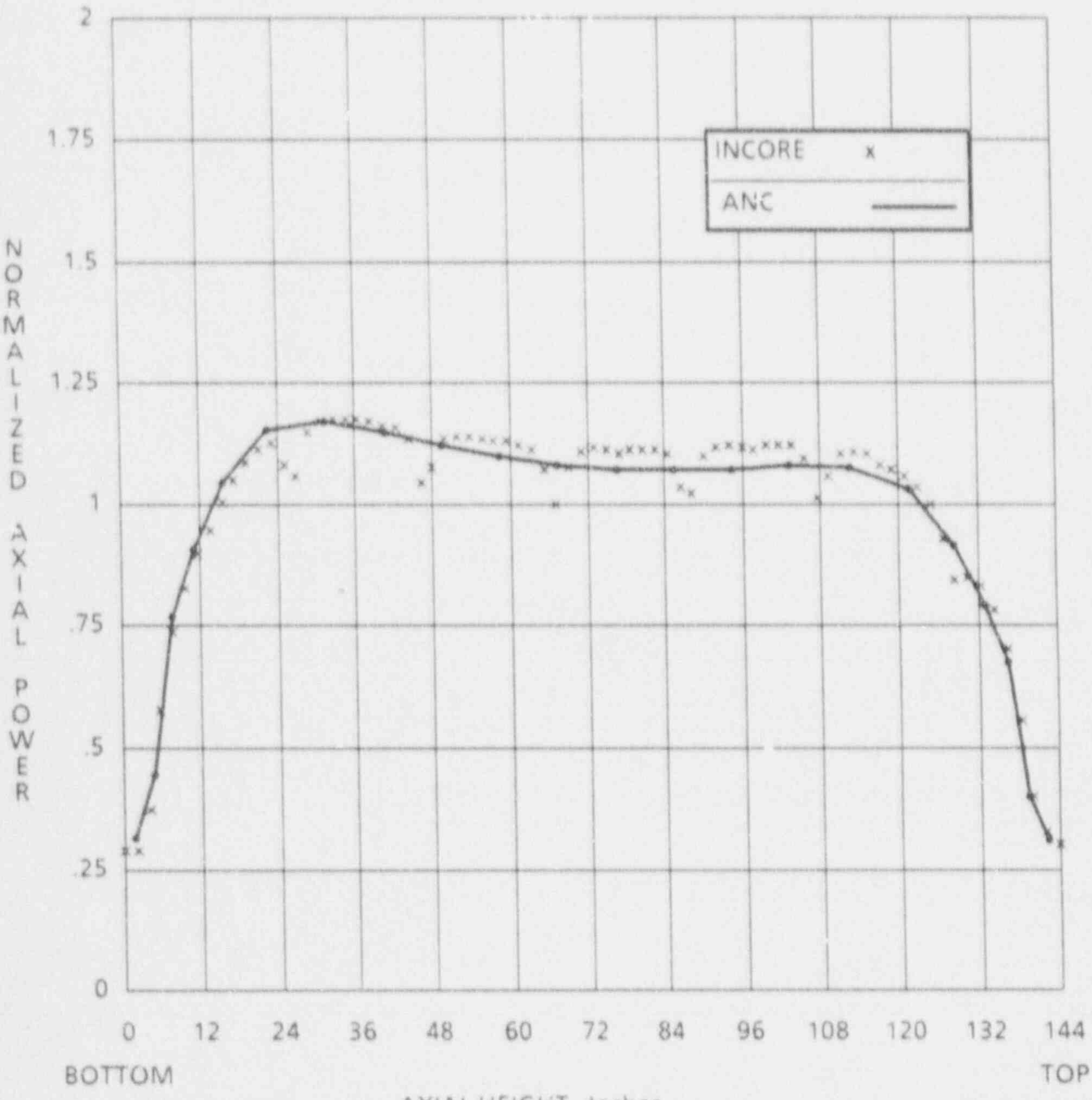
V.C. SUMMER NUCLEAR STATION CYCLE 5
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-05-015



BURNUP = 6174 MWD/MTU POWER LEVEL = 100.0% D BANK AT 230 STEPS

FIGURE 4.3-43

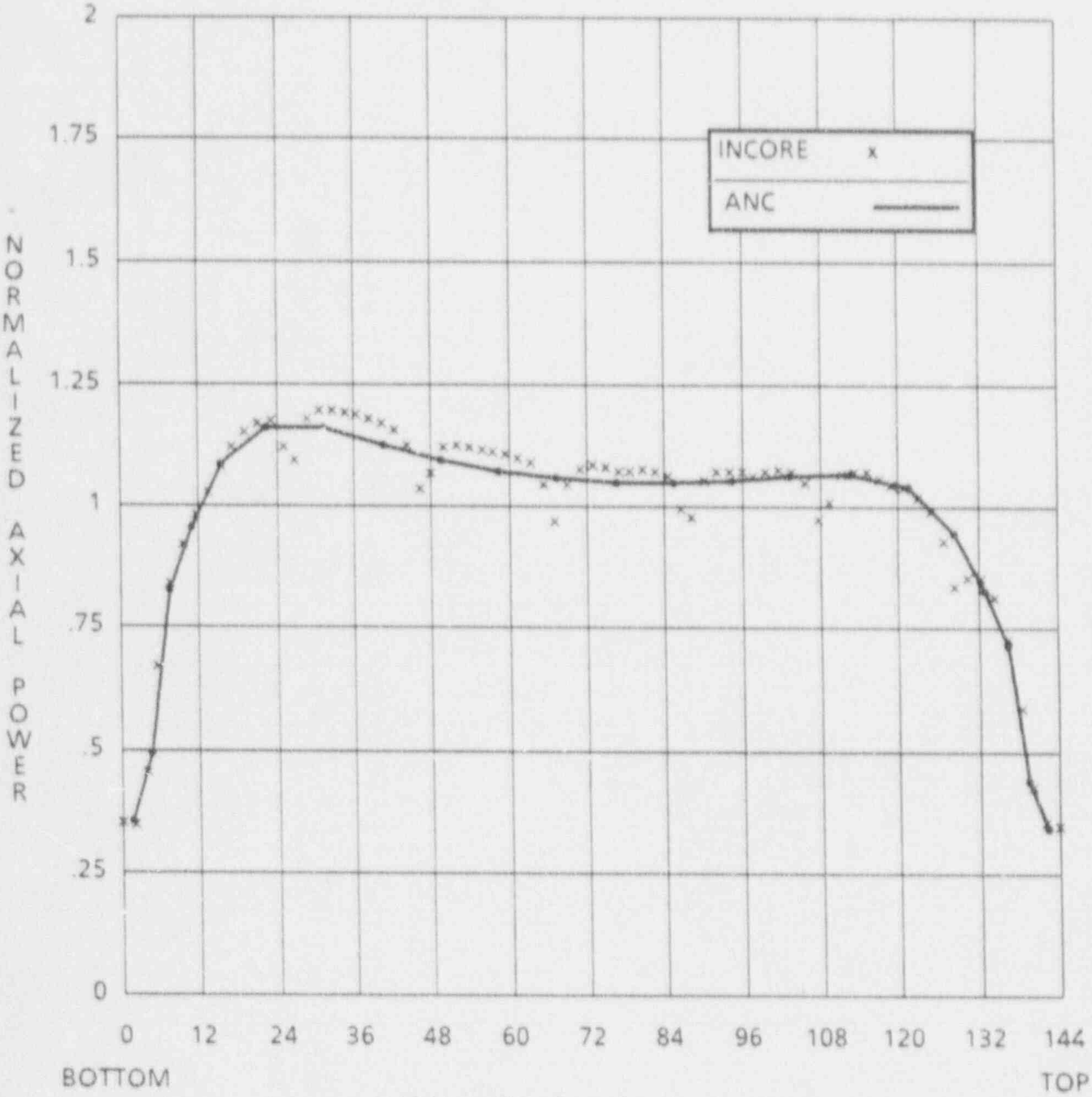
V.C. SUMMER NUCLEAR STATION CYCLE 5
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-+CFM-05-018



BURNUP = 8163 MWD/MTU POWER LEVEL = 99.8% D BANK AT 230 STEPS

FIGURE 4.3-44

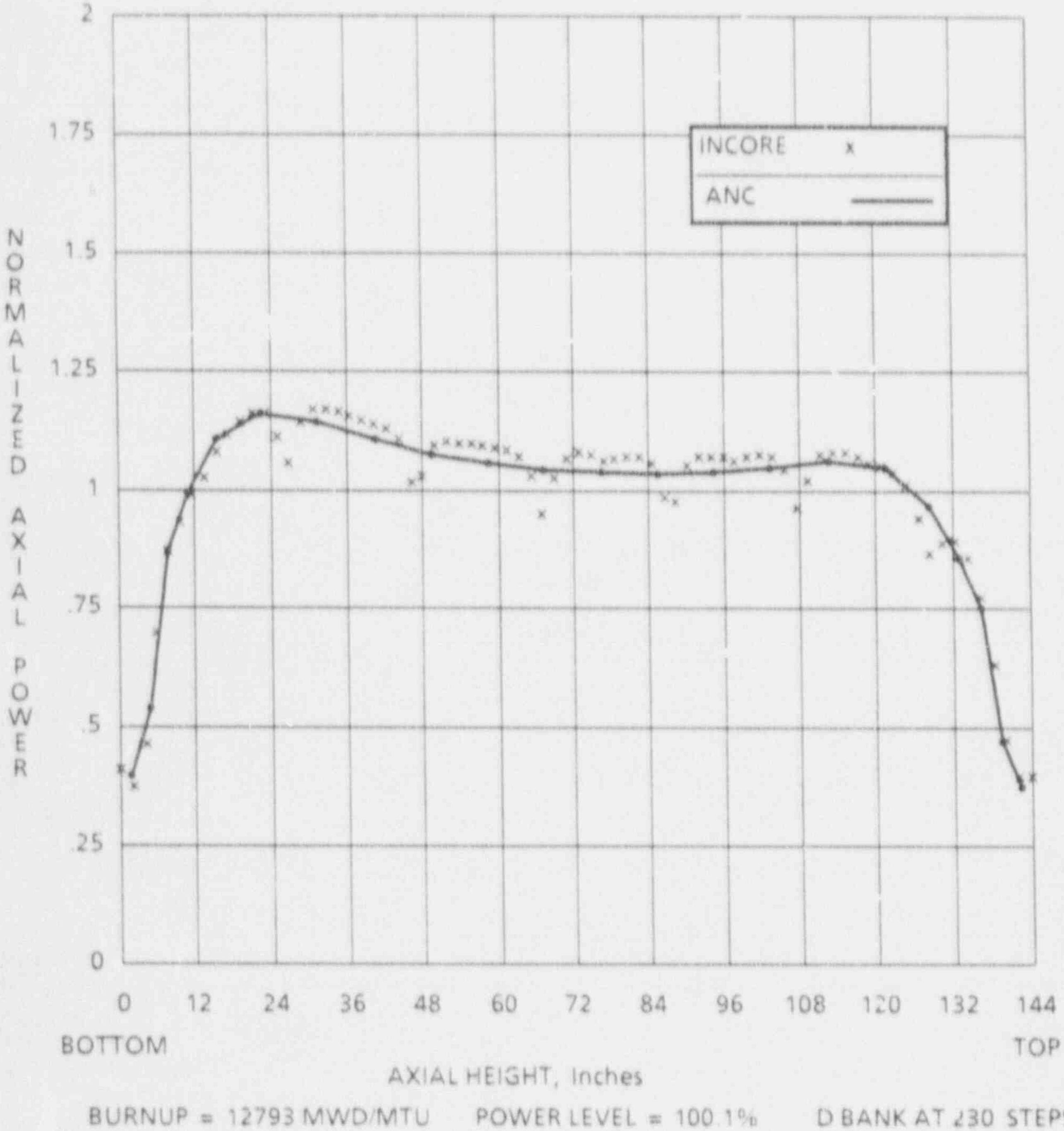
V.C. SUMMER NUCLEAR STATION CYCLE 5
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-05-020



BURNUP = 10539 MWD/MTU POWER LEVEL = 99.9% D BANK AT 230 STEPS

FIGURE 4.3-45

V.C. SUMMER NUCLEAR STATION CYCLE 5
AXIAL POWER DISTRIBUTION COMPARISON
BETWEEN INCORE AND ANC
FOR MAP-FCFM-05-022



BURNUP = 12793 MWD/MTU POWER LEVEL = 100.1% D BANK AT 230 STEPS

"This page intentionally blank."

5.0 REFERENCES

1. Langford, F. L. and Nath, R. J., "Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7308-L, April 1969, and Spier, E. M. and Nguyen, T. G., "Update to WCAP-7308-L (Proprietary), Evaluation of Nuclear Hot Channel Factor Uncertainties," March 1984.

2. Meyer, C. E. and Stover, R. L., "INCORE Power Distribution Determination in Westinghouse Pressurized Water Reactors," WCAP-8498, July 1975.

3. Nguyen, T. Q., et al, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A (Proprietary), November 1987.

4. Miller, P. W., et al, "Relaxation of Constant Axial Offset Control/FQ Surveillance Technical Specification," WCAP-10216-P-A (Proprietary), June 1983.

5. Bordelon, F. M., et al, "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A (Proprietary), July 1985.

6. Camden, T. M., et al, "Rod Bank Worth Measurements Utilizing Bank Exchange," WCAP-9863-A (Proprietary), May 1982.

7. Camden, T. M., et al, "PALADON-Westinghouse Nodal Computer Program," WCAP-9485 (Proprietary) and WCAP-9486, December 1978 and Supplement 1, WCAP-9485-A (Proprietary) and WCAP-9486-A (Non-Proprietary), September 1981.

8. Liu, Y. S., et al, "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A (Proprietary), December 1985.

9. Poncelet, C. G., et al, "LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS," WCAP-6073, April 1966.

10. Olhoeft, J. E., "The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements," WCAP-2048, July 1962.

11. Harris, A. J., et al, "A Description of the Nuclear Design Analysis Programs for Boiling Water Reactors," WCAP-10106-P-A (Proprietary), June 1982.

12. Barry, R. F., et al, "The PANDA Code," WCAP-7048-P-A (Proprietary) and WCAP-7757-A, January 1975.

6.0 APPENDIX

This appendix describes the major Westinghouse computer programs used by SCE&G to perform reload core design calculations for VCSNS. The codes are used in a manner similar to that described in Section 3 of Westinghouse's licensed reload methodology topical report (Reference 5). Although the codes described in this appendix are not specifically mentioned in the topical, two of the codes, FIGHTH and APOLLO (Reference 12), contain the same fundamental methodology as the licensed versions. The "updated versions" provide engineering enhancements (e.g., larger problem size capabilities, editing improvements, and minor modeling improvements) relative to the original code versions. The updated code versions were described at a meeting between the Westinghouse Nuclear Fuel Division and the NRC Core Performance Branch in October 1984, at which time the differences between the original and updated code versions were discussed. The NRC Core Performance Branch agreed that the updated code versions were fundamentally the same as the orginal versions, employing the same fundamental solution algorithms as the original versions.

The two remaining codes, PHOENIX-P and ANC, contain significant improvements to the methodologies discussed at the 1984 meeting between Westinghouse and the NRC. PHOENIX-P is a two-dimensional multigroup lattice code which does not rely on the spatial/spectral interaction assumptions inherent in the previous methodology. ANC is an advanced version of the PALADON code (Reference 7) incorporating the nonlinear nodal expansion method, the equivalence theory for cross section homogenization, and a rod power recovery model. Topical reports (References 3 and 8) qualifying PHOENIX-P and ANC for use in reload core design have been approved by the NRC.

6.1 FIGHTH

The FIGHTH code calculates effective temperatures in low enriched, sintered UO2 fuel rods for specified values of burnup, linear heat generation rate, moderator temperature and flow rate. The resultant fuel and clad temperatures are input to the PHOENIX-P code. The FIGHTH model accounts for the radial variation of heat generation rate, thermal conductivity, and thermal expansion in the fuel pellet; elastic deflection in the cladding; and a pellet-clad gap conductance which depends on the kind of initial fill gas, the hot open gap dimensions, and the fraction of the pellet circumference over which the gap is effectively closed due to pellet cracking. References 9 and 10 provide a description of the basis of the FIGHTH program.

6.2 PHOENIX-P

The PHOENIX-P computer program is a two-dimensional multigroup transport theory code used to calculate lattice physics parameters for PWR core modeling. In PHEONIX-P, the solution for the detailed spatial flux and energy distribution is divided into two major steps. In the first step, a two-dimensional fine energy group nodal solution is obtained which couples individual subcell regions (pellet, clad, and moderator) as well as surrounding pins. PHOENIX-P uses a method based on the Carlvik's collision probability approach and heterogeneous response fluxes which preserves the heterogeneity of the pin cells and their surroundings. The nodal solution provides an accurate and detailed local flux distribution which is then used to spatially homogenize the pin cells to fewer groups.

The second step in the solution process solves for the angular flux distribution using a standard S^4 discrete ordinates calculation. This step is based on the group-collapsed and homogenized cross sections obtained from the first step of the solution. The S^4 fluxes are then used to normalize the detailed spatial and energy nodal fluxes. The normalized nodal fluxes are used to compute reaction rates and power distributions and to deplete the fuel and burnable absorbers. A standard B1 calculation is employed to evaluate the fundamental mode critical spectrum and to provide an improved fast diffusion coefficient for the core spatial codes.

The PHOENIX-P code employs a 42 energy group library which has been derived mainly from ENDF/B-V files. The PHOENIX-P cross section library was designed to properly capture integral properties of the multi-group data during group collapse, enabling proper modeling of important resonance parameters. The library contains all neutronic data necessary for modeling fuel, fission products, cladding and structural, coolant, and control/burnable absorber materials present in PWRs.

Detailed discussions of the methodology and models incorporated in PHOENIX-P are contained in References 3 and 11.

6.3 ANC

ANC is a multidimensional nodal analysis program used to predict core reactivity parameters, power distributions, detector thimble fluxes, and other relevant core characteristics. In ANC, the nodal expansion method is used to solve the two-group diffusion equations. With this method, the partial currents and average neutron fluxes for a node are determined from continuous homogeneous neutron flux profiles described by fourth order polynomial expansions for each of the x, y, and z directions across the node. Discontinuity factors are used to modify the homogeneous cross-sections to preserve the node surface fluxes and currents that would be obtained from an equivalent heterogeneous model. ANC also contains a pin-power recovery model which couples the analytic solution to the two-group diffusion equations with pin power information from PHOENIX-P. ANC accurately reconstructs the results of fine mesh models using these methods. Reference 8 provides a detailed description of the methodology used in ANC.

ANC can be used to perform two or three-dimensional calculations with a wide variety of options. Geometries ranging from full core to octant are available with various symmetries. Feedback adjustments to macroscopic cross sections account for changes in fuel temperature and moderator density. Fuel and burnable absorber depletion and xenon and samarium buildup and decay are modeled. Typical applications of the ANC code include the determination of:

- Axial and radial power distributions,

- Differential and intergral control rod worths,
- Core reactivity coefficients,
- Critical core configurations and shutdown margins, and
- Fuel and burnable absorber loading patterns.

6.4 APOLLO

APOLLO is a one-dimensional two-group steady state diffusion theory program. An APOLLO model is normally generated by radially homogenizing a three-dimensional ANC model. Cross sections are flux and volume weighted and a burnup and elevation dependent radial buckling search is performed to normalize the APOLLO model to ANC. Since a relatively large number of mesh points are permitted in APOLLO, it is used for applications which require a finer mesh in the axial direction. Typical applications include providing:

- Axial power distributions for Fq synthesis,
- Differential and integral control rod worths,
- Trip reactivity curves,
- Load follow capability evaluations, and
- Control rod insertion limits.

Space dependent feedback effects due to xenon, samarium, rod position, boron, fuel temperature, and water density are accounted for in the calculations. APOLLO is an advanced version of the PANDA code, which is described in Reference 12.