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FEASIBILITY OF DEVELOPING SIMPLE CALCULATIONAL PROCEDURES FOR CHECKING HEAVY-ISOTOPE CONTENT IN OPERATING FUEL ELEMENTS

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

BY

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Approved by: _____

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J. R. Dietrich, Project Director

NUCLEAR DIVISION
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25

TABLE OF CONTENTS

	<u>Page No.</u>
I. INTRODUCTION	1
II. SUMMARY	5
III. APPLICATION OF PROPOSED ISOCHECK METHOD	6
IV. REACTOR DESIGN AND OPERATING DATA REQUIRED TO DEVELOP METHOD	9
A. Detail Description of Core	9
B. Reactor Operating Conditions	12
C. Anticipated Mode of Operation for Reactors with no Prior Operating History	12
D. Reactor Operating History to Time of Analysis	13
V. STEPS REQUIRED FOR SETUP OF PROPOSED METHOD	15
A. Generation of Basic Core Constants and Effective Cross Sections	15
B. Determination of Beginning-of-Life k_{∞} and M^2 for Each Fuel Type	16
C. Detailed Determination of Beginning-of-Life Assemblywise Power Distribution	16
D. Determination of FLARE Adjustable Parameters	17
E. Determination of Variation of k_{∞} and Isotopic Composition as a Function of Exposure	17
F. Preparation of Isotopic-Composition Tables or Charts	18
G. Selection of Calculation Starting Time in the Prior History of Reactor Operation and Estimation of Assemblywise Exposures Prevailing at Calculation Starting Time	18
H. Calculation with FLARE of Exposure Condition of Core as of Time of Setup of the Method	19
I. Analysis of Future Core Operations	19
VI. GENERAL APPLICABILITY OF PROPOSED METHOD	20
REFERENCES	22

I. INTRODUCTION

In view of the large quantities of nuclear fuels involved in the operation of commercial power reactors, the development of an effective system of nuclear materials management requires that methods be available for verifying the nuclear loss and production occurring in operating power reactors. In the absence of satisfactory nondestructive measurement techniques, it is necessary to rely directly on calculations for determining the heavy-isotope content in operating fuel elements. The accuracy obtainable with the calculational approach is limited to that characteristic of the state of the art of reactor physics.

Certain basic requirements should be satisfied in the choice of the calculational approach to be used for checking heavy-isotope content in operating fuel elements. These requirements are: (1) sufficient accuracy should be attainable, consistent with the limitations of the present state of the art, to permit confident and effective use of the calculational results for the required checking purposes; (2) sufficient flexibility should be inherent so as to permit reasonably straightforward application to the many diverse conditions encountered in operating power reactors; (3) where possible, reactor operating data should be utilized to the maximum extent; and (4) the computations required in the application of the calculational approach to each power reactor should be reducible to a series of well-defined and simple routine steps.

Generally, the reactor core is not discharged as a whole but instead batches of fuel assemblies are discharged from time to time. The batches are not necessarily rigorously defined

and may contain different combinations of core assemblies at different discharges. Hence, for maximum flexibility of fuel accounting procedures, the calculations should permit determination of the exposure and associated heavy-isotopic content of each individual fuel assembly constituting the reactor core.

Numerous calculational techniques are available which vary considerably with regard to complexity and cost of calculations required. It has been the practice in reactor design to simplify the calculations by reducing the dimensionality of the actual three-dimensional core geometry while retaining the use of multigroup diffusion theory. The most effective geometrical simplification for reducing the computational effort to a practical level is cylindricization of the reactor core. However, with the reactor cylindricized, the individual fuel assemblies lose their respective identity because of the geometrical distortion introduced. Hence, if fuel exposures and isotopic compositions are to be determined on a fuel-assembly basis, advantage cannot be taken of the cylindricization assumption. An alternate approach is to retain the three-dimensional geometry and to simplify the calculational problem by using only a single neutron energy group. The nuclear code FLARE is available for convenient and rapid performance of three-dimensional one-group calculations of assemblywise power and exposure distributions (see Reference 1). In the use of FLARE, a separate set of calculations must be performed, usually with a zero- or one-dimensional multigroup fuel-depletion code, in order to provide the relationships between isotopic composition and fuel exposure. FLARE permits only a coarse mesh description of the three-dimensional geometry; the resultant lack of detail is of no disadvantage inasmuch as only assembly-average values of heavy-isotope content are needed for the desired fuel accounting.

An investigation of the accuracy of predicting relative assemblywise power and exposure distributions with FLARE was carried out in Reference 2. The results indicate that the relative power output and exposure of a single fuel assembly can be predicted with a one-standard deviation error of 10% or less. For a large number of fuel assemblies in the discharge batch, the error of prediction for the batch as a whole is greatly reduced. For Yankee Core I, for which a significant amount of experimental data is available, the end-of-life exposure distribution is predicted with a one-standard deviation error of 4.7%. Reference 2 also includes a study of the applicability of the zero-dimensional multigroup fuel-depletion calculational model for relating isotopic composition to fuel exposure. Comparisons between calculational results and measured data (as averaged to smooth out the data scatter) for fuel samples analyzed from Yankee Core I show that: (1) at 14,000 MWD/MTU, the U235 depletion is overpredicted by 2.5% and the plutonium buildup is underpredicted by less than 1%; and (2) at 23,500 MWD/MTU, the U235 depletion is overpredicted by 5% and the plutonium buildup is underpredicted by 7%.

Reference 2 indicates that general prediction accuracies obtainable with present calculational techniques amount to $\pm 5\%$ and $\pm 8\%$ on U235 depletion and total Pu content, respectively, for normal batches of operating fuel elements in the well-established reactor types--the boiling and pressurized light-water reactors; this also appears to be the degree of accuracy obtainable with the combination of the FLARE code and the zero-dimensional multigroup fuel-depletion calculational model. Although the calculational approach could only be checked out for the light-water reactor, for which significant experimental data are available, the procedures are basically applicable to the other reactor types (possibly excluding the fast reactor for which further evaluation study is required). Nearly the same degree of accuracy should be obtainable

for reactors having thermal or near-thermal neutron spectra as indicated for the light-water reactors; however, no pertinent experimental data are as yet available to conclusively assure that this will definitely be the case.

On the basis of the results of Reference 2, it is concluded that calculations, which utilize FLARE for the exposure-distribution portion of the calculation and a zero-dimensional multigroup fuel-depletion model for the isotopic composition portion of the calculation, can be set up to meet the basic requirements for checking heavy-isotope content in operating fuel elements. This calculational approach is herein referred to as the ISO-CHECK method. This report presents the results of work performed at Combustion Engineering Nuclear Division for the U. S. Atomic Energy Commission under Contract No. AT(38-1)-207, Task Agreement No. VII, Phase II for the purpose of defining the specific steps and associated problems involved in the setup of the ISO-CHECK method. The work is an extension of the more general considerations of Reference 2, as related specifically to the possible development of simple methods for routinely verifying the nuclear loss and production occurring in the various power reactors, and was essentially conducted during the period April 1, 1965 through September 30, 1965.

II. SUMMARY

The specific steps involved in the setup and use of a calculational approach (herein referred to as the ISO-CHECK method) for independent verification by the U. S. Atomic Energy Commission of nuclear loss and production in commercial power reactors are described. A survey of the commercial power reactors was made to determine whether problems related to core geometrical layout and fuel, shim, and control-element management introduce difficulties which prevent essentially straightforward application of the ISO-CHECK method; it is concluded that, although each reactor requires a certain amount of special tailoring and compromises in the direct use of the method, the basic ingredients of the method can be preserved in its application to the specific reactors.

On the basis of the survey and of the results of Reference 2, it is concluded that it is technically feasible to develop a practical method for checking isotope content in reactors having thermal or near-thermal neutron spectra (which are the predominant type currently in operation and under construction). The method is sufficiently general so that it may possibly also apply to other types of reactors, e.g., the highly epithermal and fast-spectrum reactors. However, at present, it is not possible to draw definitive conclusions as to applicability to the nonthermal reactor types due to lack of pertinent experimental information. The method is amenable to updating, for improved accuracy, as operating experience is accumulated and/or improvements are made in the basic reactor physics theory; thus it may also be expected that the accuracy for the newer reactor types will be improved if and when these other types develop to the point of supporting a large through-put of fissile isotope.

III. APPLICATION OF PROPOSED ISOCHECK METHOD

Once the ISOCHECK method is set up for a given power reactor, the steps involved in its application to the given power reactor consist of:

- (1) Calculation of assembly-by-assembly exposure, by the FLARE code, for successive intervals of power operation, using the known locations of the fuel assemblies, the observed average positions of the control elements, and the recorded history of the reactor thermal power level during each interval.
- (2) Determination of assembly-by-assembly isotopic composition by applying the FLARE-calculated exposures to previously prepared tables or charts of isotopic composition versus fuel exposure.

For convenience, the input to FLARE can be categorized as being of two types. The first type consists of the reactor operating data which are observable during reactor operation and are normally recorded on a routine basis by the reactor operator. The second type consists of numerical constants (i.e., basic core constants, fitting parameters, etc.) which describe the core nuclear characteristics and the variations of these characteristics with fuel exposure, local power density, etc. These constants will have been generated in the process of setting up the ISOCHECK method and hence will be available in suitable form on computer punch cards; no changes in these constants will be required unless changes are made in the core composition, or

unless experience and/or improvements in basic reactor theory should indicate the desirability of modifying the basic reactor data to obtain improved accuracy.

The calculations for a given time interval of reactor operation also require, as input to FLARE, the exposure condition of the core as of the beginning of the time interval. This initial-condition input is actually part of the output obtained in those FLARE calculations performed for the preceding time interval and is available in suitable punch-card form for direct input in the analysis for the time interval under consideration.

For setup and use of the isotopic tables (or charts), the reactor core is coarsely zoned into a number of regions which are judged to have sufficiently different neutron spectra, on the average, so as to require separate consideration. The minimum number of such regions corresponds to the number of different fuel-assembly types contained in the core, i.e., a separate set of isotopic tables is prepared for each fuel-assembly type. Where variations occur in the amount of moderator associated with the fuel of each fuel-assembly type (e.g., due to void variations in the boiling-water reactors), additional subdivisions of each characteristic fuel-type region is required so that the resultant regions characterize not only the fuel type but also the amount of moderator relative to fuel contained in the region. For example, in a boiling-water reactor with three different types of fuel, a total of about nine regions could be involved so as to cover three different average void contents for each fuel type; in this case, nine different sets of isotopic tables would be set up and used.

The calculations for preparation of the isotopic tables are performed with a zero-dimensional multiregion fuel-depletion code (CRUMB or equivalent). A description of the calculational approach used in such zero-dimensional analyses is presented in Reference 2.

Combination of the FLARE exposures with the isotopic tables so as to obtain total isotopic content of each fuel assembly is normally a straightforward procedure. For the boiling-water reactors wherein axial variations in void content (and hence, in neutron spectra) occur, an auxiliary machine code may be required to properly combine the FLARE output with the tabulated isotopic-composition data. The isotopic tables would be incorporated as a data library in the auxiliary machine code. The code would be programmed for use by the AEC to perform the following operations:

- (1) Directly accept FLARE output on axial variations of exposure and void content along the length of each fuel assembly.
- (2) Interpolate among the tabular isotopic-composition data (according to local exposure level and local void content) to obtain local isotopic composition at the various axial levels along the length of each fuel assembly.
- (3) Integrate the local isotopic compositions over the fuel-assembly length to obtain total isotopic content of each fuel assembly.
- (4) Print out assemblywise total isotopic contents.

IV. REACTOR DESIGN AND OPERATING DATA REQUIRED TO DEVELOP METHOD

The work of Reference 2 indicates that, in principle, a method of approach is available which represents an optimum combination of simplicity, generality, and accuracy for rapid checking of fuel exposures and isotopic compositions in thermal power reactors. However, many practical problems arise in the setup of the procedures and these problems require consideration in order to properly assess practicality.

Two general areas of calculations are involved, namely: (1) the determination of the relationships between fuel isotopic composition and exposure corresponding to the neutron spectra prevailing over the various regions of the reactor core; and (2) the determination of assemblywise exposure from input information on reactor operating history. Distinction must be made between the work and information required to set up the method for each reactor and the procedures involved in the actual use of the developed method. The latter procedures can be rigorously defined in a straightforward manner utilizing only operating data on reactor power generation, fuel assembly locations, and average control-element positions. However, setup of the method for each reactor requires considerably more information pertinent to the design and operating conditions of the reactor; it is the problems associated with the acquisition and use of these design and operating data which are of special concern herein.

A. Detail Description of Core

It is important to define the core geometry and the detailed compositions of the various core regions as accurately as possible. The most effective way of accomplishing this is by means

of a series of drawings for each reactor (herein referred to as reactor data sheets). The initial information on the reactor cores can be obtained from Hazards Summary Reports submitted to the U. S. Atomic Energy Commission by the reactor operator as part of the license application for construction and operation of the reactor. On the basis of the preliminary information, drawings are prepared with pertinent explanations included on or appended to each drawing. The drawings should satisfy the following requirements: (1) be simple; (2) be easy to update; (3) indicate what additional information is needed; and (4) provide the starting point for projected performance of the required physics calculations. A typical set of preliminary reactor data sheets prepared for Core B of the Indian Point reactor is illustrated in Figures 1(a) through 1(f).

The next step of the process is to confirm the core description, firm up the questionable items, and fill in the missing details by discussions with the reactor designer and/or operator. The preliminary set of reactor data sheets provides the focal point for convenient acquisition of the additional information. Following this step, a set of final reactor sheets is prepared to provide the basis for setup of the ISO-CHECK method.

The particular drawings best suited for concise display of the core description and operations will vary from reactor to reactor. However, certain basic types of descriptions are common to all reactors; these are indicated as follows:

1. Basic Fuel Rod Design

This drawing gives a dimensioned cross-sectional view of the fuel rod (or rods, if several types are used) indicating the composition and density of the fissionable

and fertile material contained in the fuel, the clad composition, etc. For example, see Figure 1(b).

2. Basic Fuel Assembly Design

This drawing gives a dimensioned cross-sectional view showing the fuel-rod pitch (or pitches) and assembly-box pertinent dimensions and compositions. For example, see Figure 1(c). Pertinent structural members used to tie together the individual fuel rods within the assembly box are defined. Several different types of fuel assemblies may be used, varying in fuel-rod pitch and/or enrichment.

3. Control and Displacer Elements and Shims

This drawing gives a dimensioned cross-sectional view and compositions of the poison and follower (if present) sections of all types of control elements present. For example, see Figure 1(d). Displacer (or dummy) elements and shims are similarly described and identified.

4. Basic Core Configuration

This drawing gives a dimensioned cross-sectional view of the core (indicating also the active core height) showing: (a) overall core dimensions; (b) dimensions pertinent to the placements of the various types of fuel assemblies in the core, thereby also defining the channels between fuel assemblies; and (c) locations of control elements, displacer elements, and shims in the core. For example, see Figure 1(e).

5. Identification of Core Locations

The core locations available for insertion of fuel assemblies should be rigorously defined by appropriate identification, preferably corresponding to that used by the reactor designer and/or operator. Each fuel assembly has a serial number; hence, the location of each fuel assembly in the core at any time during operation is identified by giving the serial number together with the corresponding core-location designation. The control elements in the core should be numbered and each grouping of control elements used during operation should be identified. Displacer elements and shims should also be identified to assist in the determination of their placements in the core.

B. Reactor Operating Conditions

Information generally required is that relating to the determination of the densities of the moderator and coolant and of the Doppler-broadening reactivity effects in the fuel. These include, in addition to the full-power rating of the reactor, the pressures and temperatures of the moderator and coolant and the average temperatures within the fuel rods. For example, see Figure 1(a). For the boiling-water reactors, specific information is also required of coolant flow, coolant-inlet subcooling, etc. so as to permit proper determination of the steam-quality variations along the length of the fuel assemblies.

C. Anticipated Mode of Operation for Reactors with no Prior Operating History

The following information is of assistance to, but not crucial for, the setup of the calculational procedures.

- (1) Expected positions of the control rod groups at start of full-power operation with equilibrium xenon.
- (2) Expected shim and control-element management and refueling (including fuel switching) procedures. For example, see Figure 1(f) for a general plan of a refueling scheme.

D. Reactor Operating History to Time of Analysis

For reactors which have been in operation as of the time of setup of the ISOCHECK method, it is necessary to account for the prior operating history of the reactor. The purpose is to enable determination of the assemblywise exposures so as to provide the required starting point for future analysis of subsequent reactor operations. It is anticipated that the most detailed operational data required will consist of monthly records of accumulated thermal-energy production, fraction of full-power rating at which the major portion of the operation took place, and average positions of the control-element groups. A record of the core locations occupied by each fuel assembly, as identified by its serial number, should be obtained; the date at which each fuel assembly was removed from a given core location and placed in a new core location is required. Where long periods of operation took place at essentially uninterrupted constant-power conditions, it is anticipated that somewhat less detailed records will suffice.

Where the operation has been straightforward and does not extend over an exceedingly long period of time, the gathering and use of the operating data should present no unusual difficulties. However, in some reactors it may be difficult, to the point of impracticality, to take into account all of the early operating history. As examples, Dresden I has been in operation for a long period of time and has had many variations in core loading; in Saxton and Big Rock Point, experimental

programs have been carried out involving many complicated operations and core changes. In such cases, it may prove more practical to select some time point, as early as practical in the reactor's operating history, as the starting point for detailed consideration of subsequent reactor operations. The fuel-assembly exposures accumulated during operation prior to the chosen starting point (herein referred to as the calculation starting time) would then have to be estimated, taking into consideration, in a gross sort of way, the prior modes of operation and the operator's and/or designer's estimates of the exposure values at the calculation starting time. With estimates of assemblywise fuel exposures at the calculation starting time, the ISOCHECK method would be applied to obtain the assemblywise exposures as of the date of the analysis, thereby giving the required initial conditions for analysis of future core operations.

V. STEPS REQUIRED FOR SETUP OF PROPOSED METHOD

This section of the report describes the steps required for setup of the ISOCHECK method, beginning with the reactor design and operational data discussed in Section IV. A flow chart identifying the steps and their interrelationship is presented in Figure 2.

A. Generation of Basic Core Constants and Effective Cross Sections

The one-group constants used in the FLARE calculations and the relationships between isotopic composition and exposure are based on multigroup analyses. The library of nuclear parameters and microscopic cross sections as a function of neutron energy for U238-U235 systems is that used in Reference 2 for analysis of Yankee Core I. A comparable library for thorium systems is available which, on the basis of the heavy-isotope cross section data comparisons made in Reference 2, is in reasonably good agreement with those used by the various reactor designers.

For each region of the subdivided core considered for the overall analyses (method of subdivision indicated in Section III), a unit lattice cell is chosen consisting of fuel surrounded by moderator and structure so as to appropriately characterize the average compositional makeup of the region. FORM or GAM is then used to provide the spectral averaging of the basic nuclear constants over each above-thermal energy group utilizing appropriately derived L-factors. For the thermal group, TEMPEST is used to determine the Wigner-Wilkins spectrum for the volume-weighted composition of the homogenized unit lattice cell; the thermal disadvantage factor is then obtained from a cylindrical-geometry P-3 calculation of the heterogeneous unit cell using thermal-neutron constants appropriate to the Wigner-Wilkins spectrum. The Wigner-Wilkins

spectrum and thermal disadvantage factor provide the basis for obtaining the effective thermal cross sections. Blackness theory is used to obtain the nuclear constants for control regions.

B. Determination of Beginning-of-Life k_{∞} and M^2 for Each Fuel Type

The beginning-of-life values of infinite multiplication factor, k_{∞} , and migration area, M^2 , are required for use in the FLARE calculations. Values are required for each fuel type with and without adjacent control elements and for a range of void conditions encountered (in the case of the boiling-water reactor). For this determination, a fuel-assembly cell of the reactor core is chosen, which consists of the fuel assembly itself and associated surrounding media (e.g., the associated water channels in the case of the water reactors). Each composition region of the cell (e.g., fuel region, assembly-box structure, water channels, control regions) is discretely represented and nuclear constants appropriate to each region are obtained. Cell PDQ calculations (or, if deemed equally appropriate, one-dimensional radial calculations) are then performed to obtain k_{∞} and M^2 . Rodded and unrodded conditions, and a range of void conditions for the case of the boiling-water reactors, will be included in the investigation for each fuel type.

C. Detailed Determination of Beginning-of-Life Assemblywise Power Distribution

FLARE involves the use of adjustable parameters which are normally chosen to result in best determination of the assemblywise power distribution at beginning-of-life conditions. The adjustable parameters thus chosen are used throughout the burnup. Where pertinent experimental data are lacking, the choice of the FLARE adjustable parameters is made on the basis of obtaining best agreement with the results of more precise and detailed two-dimensional PDQ calculations on assemblywise power distributions. One or more such detailed calculations may be performed for a

core cross section to include the important control-element configurations used during operation. In general, the various composition regions of the core cross section are separately considered including, where possible, the individual fuel-rod lattice cells. As a result, the PDQ calculations performed often give rod-by-rod power distributions in addition to the relative assembly power outputs.

D. Determination of FLARE Adjustable Parameters

FLARE calculations corresponding to the two-dimensional PDQ calculations referred to in Step C are performed over a range of values of the FLARE adjustable parameters. The values of k_{∞} and M^2 inputted into the FLARE calculations are those obtained in Step B. The choice of the FLARE adjustable parameters (i.e., type of transport kernel, reflector albedos, a , and mixing parameter, g) is then made on the basis of obtaining best overall agreement with the detailed PDQ calculational results on assemblywise power distribution.

E. Determination of Variation of k_{∞} and Isotopic Composition as a Function of Exposure

The variation of k_{∞} with exposure and void content for each fuel type must be specified as input to FLARE in order to determine the assemblywise exposures accumulated during reactor operation. This input information is generated by performance of separate calculations which determine the variations obtained in the composition of the various core regions (specified as indicated in Section III) as a function of fuel exposure. The important compositional variations are of the heavy-isotopes, fission-product aggregates, and burnable poisons (where incorporated in the reactor design). The separate calculations are performed by means of a zero-dimensional multigroup fuel-depletion code which gives directly the heavy-isotope composition and the k_{∞}

variation as a function of fuel exposure. The basic core constants and effective cross sections, derived as indicated in Step A, are utilized in the zero-dimensional calculations. In the case of boiling-water reactors, a steam volume fraction versus quality relation is also input to the calculations. Several approximate expressions are available which fit the experimentally observed correlation between voids and quality.

F. Preparation of Isotopic-Composition Tables or Charts

From the results obtained in Step E, isotope tables or charts are prepared relating isotopic composition to fuel exposure for each fuel type and void condition.

G. Selection of Calculation Starting Time in the Prior History of Reactor Operation and Estimation of Assemblywise Exposures Prevailing at Calculation Starting Time

For those reactors which have only recently been put into operation and for which the operations have been straightforward, the calculation starting time will correspond to the beginning of reactor power operation, at which time zero-exposure conditions prevail for all fuel assemblies in the core. For reactors under construction, the calculation starting time will obviously also correspond to the beginning-of-life condition. As indicated in Section IV-D, it may be necessary, in the case of several reactors, to select a calculation starting time which does not correspond to beginning-of-life. In this case, it will be necessary to obtain an estimate of the assemblywise exposures accumulated during those reactor operations which occurred from the beginning-of-life to the selected calculation starting time. The specific way in which the estimates are made will depend on the information available from the reactor designer and/or operator. It is possible that FLARE calculations could be performed which only very roughly approximate the complicated history of reactor operation up to the calculation starting time so as

to obtain rough estimates of the fuel exposures prevailing at the calculation starting time. In every case, the calculation starting time will be chosen as early in the history of reactor operation as practical so that the errors introduced by the initial exposure estimates will be minimized in subsequent analysis of future core operations.

H. Calculation with FLARE of Exposure Condition of Core as of Time of Setup of the Method

Starting with the calculation starting time (generally, beginning-of-life) and the exposure conditions prevailing at the calculation starting time (generally, zero exposure), the detailed history of reactor power generation, control-element movement, fuel rearrangement, and refueling is considered. FLARE is utilized to determine the assemblywise exposures prevailing at the time of setup of the ISO-CHECK method; the results obtained give the initial conditions for further following the exposure variations during future reactor operations.

I. Analysis of Future Core Operations

Upon completion of steps A through H, sufficient information will be available for straightforward following of future core operations with a minimum of calculational effort. As a consequence of the setup work, computer punch cards will be available containing the pertinent core constants, fitting parameters, and the assemblywise exposure values prevailing at the time of setup of the method. Hence, only the reactor operating history from the time of setup of the method will be required as input information for analysis of future core operations.

VI. GENERAL APPLICABILITY OF PROPOSED METHOD

A general survey was made of the design features, operating conditions and fuel and control-element management schemes for the commercial power reactors of interest in order to ascertain specific applicability of the proposed method to each reactor. It was considered especially important to determine whether sufficient flexibility is inherent in the present version of FLARE for adequate handling of the many diverse conditions encountered during operation of the many reactors of different designs.

It was concluded from the survey that a certain amount of special tailoring and compromise is required to adapt FLARE to some of the reactors inasmuch as the geometrical patterns and compositional variations treated by FLARE do not precisely match those encountered in every reactor design. For example, there is no provision in FLARE to treat, in proper geometrical perspective, the presence of axially nonuniform shims or to handle axial changes in fuel-rod pitch as encountered in several of the power reactors (e.g., BONUS and Elk River). Hence, for these reactors, axial simplifications are required in the use of FLARE; these simplifications can be tolerated inasmuch as mainly assemblywise averages of the accumulated exposures are required and inasmuch as the average radial power distribution is not sensitively dependent on the axial power distribution. In a number of the power reactors (e.g., Piqua and CVTR), the fuel assemblies are arranged in a hexagonal pattern whereas FLARE is set up to handle square arrays of fuel assemblies. Hence, in these cases, some geometrical distortion of the actual core arrays is required to permit use of FLARE. Because of the control-element management scheme used in the Dresden reactor, a half-core representation of the reactor must be analyzed; hence, it is not

possible to treat each fuel assembly individually in the FLARE calculations but rather a four-assembly cluster must be taken as the basic unit for analysis.

In spite of these and other minor compromises involved in the use of FLARE, the required geometrical deviations from exact representation are not nearly as severe as encountered in the usual design calculations where one- and two-dimensional analytical models are employed. On this basis, it is concluded that the basic limitations of the present version of FLARE are not overly restrictive, insofar as seriously affecting the accuracy of the calculational results due to the compromises in geometrical representation required for some of the reactors.

Improvements in the calculation of isotopic compositions will be forthcoming as more detailed and precise nuclear data are obtained and as more experimental data from the operating reactors become available for checking the accuracy of methods of determining effective cross sections. With these improvements, updating of the method will be possible to give improved accuracy of predictions. As part of the updating, work on an expanded and somewhat more flexible version of FLARE may then be appropriate.

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Indian Point Core B

Owner:	Consolidated Edison Company of New York, Inc.		
Type:	Pressurized water	Designer:	Westinghouse Electric Corp.
Location:	Indian Point, New York	Startup:	1966
Net Power:	270 Mwe	Reactor Power:	615 Mwt

General Information

The first (thorium) core of the Indian Point Reactor is scheduled for shutdown at the end of October, 1965. The new core, first of two ordered from Westinghouse and referred to as Core B, is a three-zone low-enrichment UO_2 core. Startup is scheduled for April 1966. In the meantime, the following changes will be accomplished: refueling, installation of new silver-indium-cadmium control rods and chemical shim control, increase in the primary and secondary flow and an increase in power from 585 to 615 Mwt.

Most of the reactor structural components will be the same as for Core A. The core is housed in a pressure vessel with inside dimensions: diameter 9'9", height 36'9". The minimum wall thickness is 6.95". The internal structure consists of the lower grid plate assembly, the core shroud and thermal shields, the upper grid-plate assembly and the hold-down columns. These components are fabricated of Type 304 stainless steel. The hold-down columns will be replaced by columns of a different design so as to provide additional restriction of bypass flow. The only additional change in the core support structure will be the insertion of stainless steel bars in baffle spaces around the core to restrict bypass flow. There is no in-core instrumentation.

Reactor Operating Conditions

Operating pressure	1535 psig
Average coolant temperature	497.5 °F
Average fuel temperature	1400°F
Average neutron flux; fast	$1.9 \cdot 10^{14} \text{ cm}^{-2} \text{ sec}^{-1}$
thermal	$2.7 \cdot 10^{13} \text{ cm}^{-2} \text{ sec}^{-1}$
Initial conversion ratio	0.50
Buckling, B^2	$10.74 \cdot 10^{-4}$
$\text{H}_2\text{O}/\text{U}$ volume ratio	3.25
Effective multiplication; hot clean	1.181
hot, full power, equilibrium Xe & Sm	1.118
Total control rod worth at power	14.5%
Core loading: (UO_2)	27,200 kg
Full power lifetime, first cycle	9,660 hr
Average burnup for first cycle	10,000 MWD/T of U
Minimum equilibrium burnup	16,500 MWD/T of U

References

- (1) USAEC Docket No. 50-3, Final Hazards Summary Report for the Consolidated Edison Indian Point Reactor Core B.
- (2) USAEC Docket No. 50-3, Conceptual Design of Core B, September 1962.
- (3) Power Reactor Technology, 6(3): 66 (June 1963).

FIGURE 1(a) REACTOR DATA SHEETS FOR INDIAN POINT REACTOR, CORE B
GENERAL INFORMATION AND REACTOR OPERATING CONDITIONS

Indian Point Core B

Standard Fuel Elements:

Pellet Diameter	0.313 in
Pellet Length	0.620 in
Clad Thickness	0.012 in

Pellet Density:

Zone I and II	10.5 g/cm ³
Zone III	10.36 g/cm ³

Type X Fuel Elements:

Pellet Diameter	0.280 in
Pellet Length	0.600 in
Clad Thickness	0.0285 in

Active Fuel Length:

Zone I and II	101.5 in
Zone III	100.3 in

Type Y Fuel Elements:

Pellet Diameter	0.297 in
Pellet Length	0.620 in
Clad Thickness	0.020 in

Enrichment:

Zone I	2.87193%
Zone II	3.26310%
Zone III	4.07611%

FIGURE 1(b) REACTOR DATA SHEETS FOR INDIAN POINT REACTOR, CORE B
FUEL ROD DESIGN

Indian Point Core B

Rods per assembly:	standard	162
	Type X	1
	Type Y	10
Rod Pitch:	Type A	0.453 x 0.453 in
	Type B	0.441 x 0.441 in
	Type C	0.441 x 0.453 in

Each fuel assembly is identified by a number engraved in two places on the top nozzle of the assembly. Loading schedule for Core B is not available at the present time.

FIGURE 1(c) REACTOR DATA SHEETS FOR INDIAN POINT REACTOR, CORE B
FUEL ASSEMBLY DESIGN

Indian Point Core B

No. of control rods	21
Control rod material	Ag-In-Cd
No. of absorber rods per control rod:	
Large diameter rods	32
Small diameter rods	12
Absorber material diameter:	
Large diameter rods	0.4045
Small diameter rods	0.3290
Thickness of SS tube:	
Large diameter rods	0.0164
Small diameter rods	0.0137
Absorber material length	99 in.

Control Rod Followers:

Length of followers	90.5 in
Material of followers	Zircaloy-2

Dummy rods:

Material:	Zircaloy-2
Shape:	4 T-shaped, 2 adjacent legs half the normal thickness
	4 T-shaped
	8 T-shaped, one leg half the normal thickness

Filler bars:

There are a total of 56 type 304 SS filler bars consisting of 14 different shapes to fit the contour of the holes within the core shroud.

Soluble poison:

Boric acid is used to control excess reactivity until there is sufficient burnup to permit hot shutdown without boron. It is used also for cold shutdowns.

At the start of the first cycle, approximately 7800 ppm boric acid (1360 ppm of natural boron) will be required in the unrodded core. The inverse worth of the boric acid at this level of poisoning is approximately 687 ppm / %Δk/k.

FIGURE 1(d) REACTOR DATA SHEETS FOR
INDIAN POINT REACTOR, CORE B
CONTROL- AND DISPLACER-ELEMENT
DESIGN

Indian Point Core B

LEGEND

- - Fuel Element, Region I
- - Fuel Element, Region 2
- - Fuel Element, Region 3
- - Hold Down Column
- - Control Rod
- - Filler Rod
- - Filler Bars

Reactor Core Data:

Active Height	101.5 in
Equivalent Core Diameter	78.15 in
No. of Fuel Assemblies	120
No. of Control Rods	21
No. of Dummy Rods	16
Total Weight of UO ₂	27,200 kg

	Volumes (cu in.)	Weights (at power) (kg)
Fuel (UO ₂)	159,400	27,600
Water	258,200	3,400
Stainless Steel	38,200	5,000
Zr	24,800	2,700
Total Void	486,900	38,700

Control Rod Group Identification:

Group	Control Rods
B	5, 7, 15, 17
C	2, 9, 13, 20
A	6, 10, 12, 16
D	1, 8, 14, 21
E	3, 4, 11, 18, 19

There are four Po-Be sources located in four region I fuel assemblies and a total of twenty Sb-Be sources located in region I and II fuel assemblies. The sources are located in place of the center fuel pin in the fuel assemblies.

FIGURE 1 (e) REACTOR DATA SHEETS FOR INDIAN POINT REACTOR, CORE B CORE LAYOUT

Indian Point Core B

ENRICHMENT, WEIGHT %

Initial Zone I	2.87193%
Initial Zone II	3.26310%
Initial Zone III	4.07611%
New Fuel	~ 4.1%

FIGURE 1(f) REACTOR DATA SHEETS FOR INDIAN POINT REACTOR, CORE B
FUEL MANAGEMENT SCHEME

FIGURE 2 FLOW CHART DEFINING STEPS REQUIRED FOR SETUP AND USE OF THE ISO-CHECK METHOD FOR EACH REACTOR