NUCLEAR FUEL PERFORMANCE EVALUATION

EPRI NP-409
(Research Project 509-1)

Final Report

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FOREWORD

The Nuclear Fuel Performance Evaluation Project (RP509) is an integral part of the Electric Power Research Institute's Fuel Performance Program. The work was structured as a fundamental building block in the EPRI plans to create and demonstrate a site-based, near-real-time, fully computerized logic system for fuel reliability tracking and predicting. To accomplish this, the RP509 project results, and those from other scoping/code-development efforts, have been factored into the Power Shape Monitoring System (PSMS) project (RP895) work scope and goals; this new three-year effort was initiated in 1977 with plans to complete the site evaluation at the Oyster Creek Boiling Water Reactor in late 1979.

This was a POSHO evaluation project, not a model development/improvement effort. Through the RP509 project, and in addition through other non-EPRI funded work, insights were gained as to model strengths and shortcomings, and Scandpower has independently carried out model upgrading during the two years required to complete this evaluation. This study indicates that the POSHO method is useful in its present form but shows sufficient error to encourage further improvement. The ultimate value of the method is its promise to improve reactor management and thereby reduce fuel pin failure and extend the fuel cycle. Accordingly, it will be utilized as the basis for the initial Fuel Reliability Evaluation Module and further evaluated within the PSMS (RP895) project.

Floyd Gelhaus
Project Manager

ABSTRACT

An evaluation has been made of the ability of Scandpower's empirical fuel performance model POSHO ("Power Shock") to predict the probability of fuel pin failures resulting from pellet-clad interaction in commercial nuclear power plants. POSHO provides an analytical method to calculate the failure probabilities associated with power level maneuvers for different fuel assembly designs. Application of the method provides a basis for risk-benefit decisions concerning operational procedures, fuel designs and fuel management strategies.

One boiling water reactor (BWR) and one pressurized water reactor (PWR) were selected for study to compare model predictions with actual failures, as determined from post irradiation examination of the fuel and activity release data. The fuel duty cycles were reconstructed from operating records and nodal power histories were created by using Scandpower's Fuel Management System - FMS computer programs. Nodal power histories, coupled with the relative pin power distribution in each node, were processed by the fuel failure prediction model, which tracks the interaction power level for each pin group in each node and calculates the power shocks and the probability for pellet-clad interaction cracks. The results of these calculations are processed statistically to give the expected number of cracks, the number of failed fuel pins in each assembly and the total number of failed assemblies in the core.

Fuel performance in the BWR, Quad Cities Unit Two, was calculated by the model in approximate agreement with the observed performance. Fuel performance in the PWR, Maine Yankee, was calculated in approximate agreement for two of the three fuel designs. The high failure rate in the third design, Type B fuel, was not calculated by the POSHO pellet-clad interaction model.

Results from a simple fission product activity release algorithm, developed as part of this project, were encouraging, but further development of this algorithm is required for it to be of practical use.

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ACRONYMS AND ABBREVIATIONS

AEC	Atomic Energy Commission
ALAP	As low as practicable (for effluent releases)
ALARA	As low as reasonably achievable (for effluent releases)
APRM	Average power range monitors (incore instrumentation for BWR's)
BOC	Beginning of cycle
BWR	Boiling water reactor
B&W	Babcock and Wilcox
CE	Combustion Engineering
CEA	Control element assembly
CRD	Control rod drive
EC	Eddy current
E-BUN	Computer routine for GE reactors to print fuel assembly (bundle) exposures
EOC	End of cycle
EPRI	Electric Power Research Institute
FDCA	Fuel Duty Cycle Analysis system of ScP
FMS	Fuel Management System of ScP
GE	General Electric
INCA	Process computer program at Maine Yankee for inferring fuel assembly radial power distribution from in-core instrumentation
KRB	Kernkraftwerk RWE-Bayenwerk (Gundremmingen)
LHGR	Linear heat generation rate
LPRM	Local power range monitors (in-core instrumentation for BWR's)
MSIV	Main Steam Line Isolation Valve
MWD/TU	Megawatt-days per metric ton of uranium
MWe	Megawatts-electrical
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
P1	A GE computer routine for printing reactor operating parameters
PC	Preconditioning (of fuel)
PCI	Pellet-clad interaction
PCIOMR	Preconditioning intermin operating management recommendations of G.E.

PCMI	Pellet-clad mechanical interaction
P-BUN	Computer routine for GE reactors to print fuel assembly (bundle) powers
PIE	Post irradiation examination (of fuel)
PINGR	Scandpower computer program for calculating the average power density for a group of fuel pins within a fuel assembly based on individual pin power distributions as calculated by the FMS program RECORD
POLGEN	Scandpower computer program for representing the pin group power distributions by polynomials as functions of exposure, void, exposure-weighted void and control fraction
PRESTO	Scandpower computer program for performing three-dimensional reactor core histories of nodal power, reactivity and flow distributions
POSHO	Scandpower model for fuel failure frequency prediction
PWR	Pressurized water reactor
RECORD	Scandpower computer program for calculating macroscopic nuclear constants, as a function of fuel burnup, void fraction and exposure-weighted void fraction, and fuel pin powers and isotopic densities
RWE	Rod withdrawal event (at Quad Cities Unit Two during Cycle 1)
SCP	Scandpower
SJAE	Steam jet air ejector
TIP	Traveling in-core probe
TUG	The Utility Group (a group of European Utilities that have sponsored the development and application of POSHO)
UT	Ultrasonic examination
VT	Videotape recording of visual examination
W	Westinghouse

Section 1

INTRODUCTION

Following earlier problems with fuel in commercial nuclear power reactors, such as quality assurance in fabrication, hydriding, and densification, which manufacturers believe now have been resolved, there still remains the statistical phenomenon of pellet-clad interaction. The phenomenon itself results from the various changes in geometry of both cladding and fuel pellets associated with pressure changes, temperature changes, linear heat generation rate changes, exposure and cracking and relocation of pellets and is affected by the embrittling effects of irradiation and the presence of fission products. The results of pellet-clad interaction (PCI) can lead to cracking and failure of the cladding and release of fission product activity to the primary coolant. Ingress of water through such cracks can lead to hydriding or increased stress-assisted corrosion cracking, with further release of activity.

To reduce the probability of cladding failures resulting from PCI, each of the manufacturers specifies limits on the rate of rise of power or linear heat generation rate. These limitations are functions of time and prior exposure. The result of these limitations on operating power level, during initial and subsequent power ascensions, is to impose a reduction in the reactor capacity factor that is reported to range from one to five per cent for boiling water reactors; somewhat less for pressurized water reactors. Even when conscientious efforts are made to follow the manufacturer's recommendations, PCI-type failures still occur. While fuel failure avoidance involves the direct cost of lost power generation, fuel failures involve costs resulting from increased radioactivity levels, increased post irradiation examination activities, fuel reassembly efforts, and increased storage and inventory requirements.

Scandpower has developed an empirical model, POSHO, for predicting the probability of PCI-type failures. The model was postulated originally on the basis of phenomenological insights gained from the experimental results reported from the OECD Halden Reactor Project. Testing and further development of the model were sponsored by a group of seven European utilities, referred to by Scandpower as The Utility Group (TUG).

Initial validation of the model was carried out under the TUG program and the model was tested against data from both European and U.S. commercial reactor experience, but at precision categories that were generally at a low level, (i.e., detailed and complete information on the fuel, its operation and performance were not available). Further development of the model and development of a larger data base, essential for statistical treatment of the PCI phenomenon, is proceeding under the second round of TUG activities, with TUG now expanded to 13 utility members in Europe.

In early 1975, Scandpower proposed to the Electric Power Research Institute (EPRI) a project for application and validation of POSHO using U.S. commercial reactor experience at the category 1 precision level (i.e., with detailed and complete information on fuel, operating history and performance). As originally conceived, this project also would have sought a very large operating experience and fuel performance data base. While EPRI was interested in validation of the model, it was thought premature to create a large data base without a better understanding of what information was essential or irrelevant for such a data base. Consequently, it was agreed that Scandpower would select one boiling water reactor and one pressurized water reactor, for which commercial operating experience data were available and suitable for evaluation of the model, and proceed to validate the model at the highest category of precision possible.

As a related effort, it was agreed that Scandpower would develop a fission product release algorithm that would relate observed fission product activity levels to the incidence of PCI-type fuel failures.

This report is an account of the work done under RP509-1, "Nuclear Fuel Performance Evaluation," which began in July 1975. It describes the bases and processes by which the boiling water reactor (BWR Quad Cities Unit Two) and the pressurized water reactor (PWR Maine Yankee) reactor selections were made, the acquisition of data, the methods of analysis, the results of the analyses and the conclusions and recommendations of Scandpower, based upon this work. Appendices are included that describe the computer programs used, that provide data sets for Quad Cities Unit Two illustrating the fuel duty cycle simulation and that provide selected results from the major events analyzed at both Quad Cities and Maine Yankee.

Section 2

SUMMARY

The calculated probabilities of fuel failure resulting from pellet-clad interaction events in two U.S. commercial nuclear power plants were compared with actual failures. The calculations were made using Scandpower's empirical model, POSHO. Actual failures were determined from post irradiation examinations of the fuel, or deduced from observed activity release data.

To carry out these studies, one BWR (Quad Cities Unit Two) and one PWR (Maine Yankee) were selected. Selection criteria included: a minimum of two fuel exposure cycles, reactor and fuel of current design, operation of the fuel at or near design linear heat generation rates, normal power maneuvering experience, availability of fuel characterization data, global and local power history data, activity release data and results of post irradiation examinations.

Fuel and reactor design data were obtained from the utility or other sources. From site visits data were obtained on either offgas activity (Quad Cities) or primary coolant activity (Maine Yankee), post irradiation examinations of the fuel and detailed operating histories.

The actual total reactor power histories were examined to identify events that could have caused power shocks to the fuel. A simplified reactor power history, which reproduced the power shocks for these events, was then constructed and simulated with the 3 - dimensional core simulator, PRESTO. Scandpower's FMS program RECORD was used to provide all nuclear parameters and cross sections used in PRESTO and relative fuel pin axial power distributions in the fuel for control rods inserted or withdrawn as a function of void (BWR only) and burnup.

The fuel duty cycles were reconstructed from operating data and nodal power histories created using Scandpower's Fuel Management System - FMS. Nodal power histories, coupled with the relative pin power distribution in each node, were processed by the fuel failure prediction model, which tracks the interaction power

level for each pin group in each node and calculates the power shocks and the probability for pellet-clad interaction cracks. The results of these calculations are processed statistically to give the expected number of cracks, the number of failed fuel pins in each assembly and the total number of failed assemblies in the core.

For Quad Cities, the gross core failure predictions were 34% high for Cycle 1 and 36% low for Cycle 2. The predicted number for Cycle 1 was 101, with 74 determined to be failed by ex-core sipping. The predicted number for Cycle 2 was 60, with 94 determined to be failed by in-core sipping. In Cycle 1 almost every fuel assembly experienced a considerable number of relatively small shocks so that the general background failure probability level was high, but there were no individual assemblies with outstandingly high failure probabilities - with one notable exception. The exception was the fuel assemblies adjacent to a control rod that was improperly withdrawn. Other than the exception, it was possible to predict the locations of fuel failures in Cycle 1 only in a general way. In Cycle 2 the bulk of the failures was caused by a few large shocks, rather than many small ones, so that geographic correlation between prediction and observation is quite good. Predictions of the offgas activity level were very good for the beginning of Cycle 1, but were too low for the end of Cycle 1 and during Cycle 2. A number of factors may explain this discrepancy - the lack of a "damage accumulation" factor in POSHO, the lack of a burnup dependency in the release model, and other factors that would tend to increase the escape rate coefficients and give higher release rates for the same failure rate.

For Maine Yankee the predicted total number of failed assemblies for the two fuel cycles was in good agreement with the sipping results for fuel assembly Types A & C. The prediction for Type B fuel was nearly the same as for Type A fuel, but the sipping results indicate nearly 10 times as many B assemblies actually failed. The B fuel was identical in design to the A fuel insofar as the POSHO model is concerned. The predicted number of failures in A & C fuel for both cycles was 7.7, the observed number of failures was 7 (48 B fuel failures were observed). The discrete behavior of the Type B fuel could be described by a pellet-clad interaction failure mechanism with an exponential power level dependence, which would be inconsistent with the current POSHO model. The details of extensive analyses of the type B fuel performance may be found in Reference (17).

The power shocks to the fuel in Maine Yankee resulted generally from gross power variations and xenon distribution changes. The magnitude of the calculated assembly failure probabilities is 5% or less, so that the location of the assemblies that actually failed could not be predicted. If the actual failures were the result of a combination of statistically distributed factors, a random distribution of their locations would be expected. The failed assemblies did, in fact, appear to be randomly located.

The predicted primary coolant activity levels differed from measured values by a factor of 25, because of the large actual number of B-type failures. Scandpower calculations would indicate not more than one pin failed per assembly. Post Irradiation examination at the end of Cycle 1 of failed A and C-type assemblies revealed one leaking rod in each. In the two B-type assemblies examined, one had 4 and the other had 11 failed pins.

This study represents the first comprehensive application of the fuel failure model, POSHO, based on detailed nodal power histories. The fuel performance calculated by the model was in approximate agreement with the actual performance with the exception of the Type B fuel in Maine Yankee. The simple model for prediction of fission product activity release gave results that could be correlated generally with the calculated fuel failure events, but did not include many factors that are considered relevant.

For further development and validation of the fuel failure prediction model, the most important activities would be the acquisition of fuel failure data (post irradiation information on failed pins in both PWRs and BWRs) and detailed nodal power histories.

For beneficial use of the fuel failure prediction technology, the most important activities would be 1), development of an on-line "shock" monitor, that would alert the utility operator to the possibility of fuel failures resulting from pellet-clad interaction, either in a real-time monitoring mode, or in a predictive mode (EPRI's project RP 895, Power Shape Monitoring System, is directed at exactly this goal), and 2), development of a more sophisticated and realistic fission product release algorithm.

Section 3

SELECTION OF PARTICIPATING UNITS

The objective of the first phase of this project was to select the reactors and fuel cycles to be used for evaluation of the Scandpower model for prediction of fuel failures resulting from power shock events. This selection process was conducted during the Summer and Fall of 1975. Two fuel cycles at a PWR and two cycles at a BWR were to be selected. The following general criteria were used in selecting reactors:

- At least one cycle of operation was to be completed by the Fall of 1975.
- The reactor design should be similar to the current product line of its supplier.
- The utility should be willing to cooperate in this study and to provide the help necessary in the data-gathering program.
- The design and manufacturing process for most of the fuel used in the selected cycles should be such that it was relatively free of any deficiencies expected to cause failures. For example, cycles in which fuel was known to have failed largely from densification collapses or internal hydriding should be avoided.
- The nominal operating power of the reactor for the fuel cycles selected should be high enough to produce average and peak linear heat generation rates (LHGRs) in the range in which pellet clad interaction (PCI) would be expected.
- The changes in local power that occurred over the course of the cycles should be typical of those expected in a normal operating cycle. (Selection of a cycle in which the fuel experienced an unusually mild duty cycle should be avoided.)
- The end of cycle exposure for the majority of the fuel should be high enough that some zirconium enbrittlement would be expected to have taken place. Reactors with only one completed cycle were to be avoided because of limited exposure and because first cycle operation tends to be atypical.
- The data from which local power and changes in local power would be calculated should be available in an easily retrievable form.
- Data on the actual fuel performance should be extensive and readily available. Both operational data (for example, steam jet air ejector offgas activity, in the BWR, and primary coolant activity in the PWR) and the results of post irradiation examination of the fuel should be available.

The reactors initially considered are listed in Table 3-1, along with their power production record. The 22 reactors with greater than 300 full power days of power production were further considered through a telephone survey which was conducted to determine the utility interest and the availability of data. Two forms were used for assistance in this survey; one for each utility and a second for each fuel cycle. Copies of these work sheets are provided as Exhibits 3-1 and 3-2. From the telephone survey 3 PWRs and 3 BWRs were identified for further study.

Visits were made to the offices of the utilities and to the reactor sites in some cases. During these visits a first-hand review of the records that would be available for this project was made. It was found that for all facilities sufficient reactor and fuel design information was available. It also was determined that the overall reactor power history data were available in the appropriate detail, but that the complete history of control rod motion at the BWRs was more difficult to obtain. The lack of complete control rod motion data for a BWR would prevent accurate calculation of changes in local power.

The availability of information on actual fuel performance varied widely. Online measurements, such as offgas activity (BWR) and primary system sample activity (PWR) were taken daily at some plants, but only weekly at others. Variations in sipping of irradiated fuel included: 100% (in- or ex-core) sipping, sipping of only the fuel meant for reinsertion and no sipping at all. Most of the PWRs, but none of the BWRs fell into this last category. A summary of the post irradiation examination programs is provided in Tables 3-2 and 3-3 for the BWRs and the PWRs, respectively.

The final selection of the BWR was influenced strongly by the availability of records that could be used to reconstruct the local power history. The necessary records appeared to have been maintained for the Quad Cities units and they were available on microfilm. The first and second cycles at Quad Cities Unit Two were chosen, because of the interesting power shock events and the other considerations listed in Table 3-4.

The final selection of the PWR was between Maine Yankee and Ginna. Table 3-5 shows a comparison of the key points in the selection process for each of these reactors. The first two cycles at Maine Yankee were chosen primarily because of the availability of superior post irradiation examination (PIE) data. Coincidentally, EPRI and Combustion Engineering had conducted a program for hot cell examination of the Maine Yankee fuel.

Table 3-1

POWER PRODUCTION

FROM

LARGE U.S. NUCLEAR POWER PLANTS

Plant Name	Number of Current Fuel Cycle	Approximate Full Power Days Through June 1, 1975
Oyster Creek	6	1359
Ginna	6	1230
Nine Mile Point - 1	5	1166
Point Beach - 1	3	1117
Robinson - 2	3	1041
Monticello	4	944
Millstone	3	911
Dresden - 2	5	811
Dresden - 3	4	720
Point Beach - 2	2	668
Quad Cities - 1	2	652
Quad Cities - 2	2	610
Turkey Point - 3	2	558
Vermont Yankee	4	524
Pilgrim - 1	2	512
Surry - 1	2	481
Maine Yankee	3	480
Turkey Point - 4	2	426
Surry - 2	2	420
Oconee - 1	2	366
Fort Calhoun	2	314
Peach Bottom - 2	1	308

Each time new fuel is added or old fuel rearranged a new fuel cycle begins.

Table 3-1 (Cont.)

POWER PRODUCTION

FROM

LARGE U.S. NUCLEAR POWER PLANTS

Plant Name	Number of Current Fuel Cycle	Approximate Full Power Days Through June 1, 1975
Palisades	1	300
Browns Ferry - 1	1	269
Indian Point - 2	1	269
Kewaunee	1	262
Three Mile Island - 1	1	262
Zion - 1	1	250
Prairie Island - 1	1	242
Cooper	1	215
Oconee - 2	1	188
Arnold	1	187
Peach Bottom - 3	1	146
Zion - 2	1	134
Prairie Island - 2	1	112
Oconee - 3	1	111
Arkansas - 1	1	107
Rancho Seco	1	103
Browns Ferry - 2	1	99
Hatch - 1	1	56
Calvert Cliffs - 1	1	55
Cook - 1	1	49
Fitzpatrick	1	12

Each time new fuel is added or old fuel rearranged a new fuel cycle begins.

TABLE 3-2 POST IRRADIATION EXAMIMATION - BWR's

BWRs Unit	Cycle No.	Sipping	Visual	EC/UT	Other Hot Cell - Profilometry
Vermont Yankee	1	100% ex core	Many bundles & many rods	Rods from 51 leakers & 8 non-leakers	None
	2	100% ex core	Some	None	None
	3	(No PIE - almost entire core replaced with 8x8 fuel)			
Dresden 2	1	Some in & ex core	1334 rods	1334 rods	None
	2	100% in core Many ex core	Rods from 39 bundles	Rods from 39 bundles	None
	3	215 ex core	Some bundles, 96 rods	96 rods	(509 suspect bundles discharged w/out exam)
	4	100%	None	None	None
Dresden 3	1	100% in and ex core	Rods from 103 bundles	4944 rods	None
	2	90% in and 10% ex core	None	None	None
	3	100% in & ex core	None	None	None
Quad Cities 1	1	100% in & ex core	None	None	None
Quad Cities 2	1	100%	Some	Some	None
	2	100%	None	None	None
Millstone 1	1	100% ex core	Some	Rods from 109 bundles	None
	2	100% ex core of fuel for cycle 3	Some	None	None
Pilgrim	1	100% ex core	None	None	None

TABLE 3-2 POST IRRADIATION EXAMINATION (Cont'd)

BWRs Unit	Cycle No.	Sipping	Visual	EC/UT	Other Hot Cell - Profilometry
Nine Mile Pt.	1	100% in core	Some	Rods from 38 bundles	None
	2	100% in core	Some	Some	None
	3	100% in core	Some	None	None
	4	100% in core	Some	None	None
Oyster Creek	1	100% in core	Some	Rods from 44 bundles	None
	2	100% in core	Some	Some GE fuel	None
	3	100% in core	Some	Some	None
	4	100% in core	Some	None	None
	5	100% in core	None	None	None

TABLE 3-3 POST IRRADIATION EXAMINATION - PWR's

PWRs Unit	Cycle No.	Sipping	Visual	Other EC/UT/Profilometry Hot Cell Exams	Notes
HB Robinson 2	1	None	100% of bundles - V.T.	None	NRC will do Hot Cell work on one assembly later
HB Robinson 2	2	None	Almost 100% " - "	Some γ scanning	
Point Beach 1	1	$\sim$ 76%	100% of bundles $\sim\frac{1}{2}$ "	None	
	2	None	Some & some V.T.	None except →	W cut one assembly
Point Beach 2	1	None	Some	None	
Surry 1	1	None	50% core some V.T.	None	
Surry 2	1	None	10 bundles V.T.	None	Removable rods on some bundles for later exam
Oconee 1	1	None	100% of bundles Some Photos	10 bundles NDT Some γ scanning	Hot cell later - 1 bundle
Maine Yankee	1	100%	Leakers only	Hot cell several rods	
Maine Yankee	2	100%	Some		
Ginna	1	Almost all	Some		
Ginna	2	Some	Most bundles & rods from 55 assemblies	None	
Ginna	3	None	None	None	Quick refueling, no PIE
Ginna	4	None	Some	None	
Ginna	5	None	Test 4 bundles	None	
Turkey Pt. 3	1	None	Some	None	
Turkey Pt. 4	1	None	Some V.T.	None	

Note: V.T. = vidiotape

TABLE 3-4 - BWR - QUAD CITIES - UNIT TWO

- EXCELLENT RECORDS SYSTEM
- TWO CYCLES COMPLETED
- DAILY OFFGAS ISOTOPIC ANALYSES
- ONLY TWO FUEL TYPES (7X7 AND 8X8)
- SHOCK EVENTS REASONABLY IDENTIFIABLE
- AVERAGE BURNUP $\sim$ 10,000 MWD/T
- 100% SIPPING AND SOME PIE
- ALL SPENT FUEL AVAILABLE FOR ADDITIONAL PIE

TABLE 3-5 - COMPARISON OF PWR CANDIDATES

DATA TYPE	MAINE YANKEE	GINNA
DESIGN AND MANUFACTURING	(NO DISCERNABLE DIFFERENCES HAVE BEEN IDENTIFIED)	
POWER HISTORY	ONE FULL CYCLE, ONE SHORT CYCLE, BOTH LIMITED TO LESS THAN FULL POWER (75-80% MOST OF TIME)	THREE CYCLES, THREE YEARS, ALL BUT 4 MONTHS AT 83% POWER OR LESS
FUEL EXPOSURE	~ 10,000 MWD/T	~25,000 MWD/T
POWER CHANGES	FIRST CYCLE - CRDs AND BORON, SECOND CYCLE - BORON ONLY	CRDs AND BORON ALL CYCLES
LOCAL POWER RECORDS	HOURLY LOGS, IN-CORE FLUX MAPS	HOURLY LOGS, IN-CORE FLUX MAPS
PIE	100% EX-CORE SIPPING, SOME ROD-BY-ROD, SOME HOT CELL	SOME VISUAL, SOME PIE ON TEST FUEL
PRIMARY COOLANT ACTIVITY		
IODINE	5 DAYS A WEEK	ONCE A WEEK
GASES	NOT ANALYZED	3 DAYS A WEEK
"LET-DOWN" MONITOR	—	YES
SUSPECTED CAUSE OF FUEL FAILURE	HYDRIDING AND PCI	UNKNOWN PCI SUSPECTED

Exhibit 3-1 Telephone Survey Work Sheet - Utility

EPRI PHASE I

Utility Name:
Address:
Phone:
Plants:

Contacts:
(Names & Titles) Ext.#

Is this utility interested in participating in the program of testing the Scandpower model?

Would they be interested in a Fuel Technology meeting to discuss the goals of the EPRI program and the ScP model, etc?

Who would attend such a meeting from this utility?

Exhibit 3-2

EPRI Phase I

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FUEL EXPERIENCE QUESTIONNAIRE

Plant ____________________

Utility ______ BWR - PWR ______ Full Power Days thru ______

MWth ______

Date of Commercial Operation ______ No. of Assemblies ______

Cycle #

Source of Information: ______

Fuel Fabricator & Type: ______

Dates of Operation: ______ through ______

1. Availability of:
   a) Fuel Design Data:

   b) Fabrication Procedure & Control Data:

   c) Power History Data:

   d) Fuel Performance Data:

      Primary System Activity or offgas data ______
      Sipping - in core/ex core (% of core) ______
      Visual Examination - Bundles ______, Rods ______
      E/C or UT - Rods ______
      Hot Cell - Number of Rods ______

2. Describe Power History - (Base Load - Load Follow)

3. Describe Local Power Control - (Follow PCIOMR's?) (CRD Motion at Power?)

4. Describe Fuel Performance (Few or Many Failures? - Cause Determined or Suspected to be - ?)

Section 4

ACQUISITION OF DATA

DEVELOPMENT OF THE QUESTIONNAIRE

After selection of the units and fuel cycles, the next phase of the work involved the collection of the fuel and reactor design data needed to predict the fuel performance and the collection of the fuel performance data necessary for comparison with the predictions. The initial task was the preparation of the composite questionnaire to be used in the data-gathering work. The questionnaire was compiled from material previously developed by Scandpower and modified on the basis of experience during the earlier data collection campaigns when the fuel failure prediction model was being tested against European and U. S. reactor experience.

The composite questionnaire, which is provided in its entirety in Appendix E, contains five main parts plus sections for figures, tables and exhibits. The five sections cover:

- The identification of fuel types and core location for each cycle,
- The design and manufacturing data for each fuel type,
- Reactor design information needed to perform the core follow calculations,
- The detailed operating history for each cycle, and
- The fuel failure and inspection information summary.

APPLICATION OF THE QUESTIONNAIRE

The questionnaire was used only as a guide in the data acquisition phase. It was sent to each of the selected utilities prior to the office and site visits so that they might be able to plan for these visits and to have some of the appropriate records readily available. The questionnaire also was used as a check list for the data collectors. Because of the vast differences in format between the questionnaire and the plant records, it was not feasible to obtain complete information for many sections of the questionnaire.

Quad Cities Unit Two

The 3 main objectives of the data-collection work during the Quad Cities site visits were to:

- Obtain the offgas history,
- Obtain the post irradiation examination records, and
- Obtain detailed information on the operating histories.

A very comprehensive record of the offgas history was obtained, including isotopic analysis of the 6 principal constituents, on an almost daily basis. These records were used in the fission product release algorithm phase of the work and also were reviewed to determine any trends that could help identify periods of the operating history where shocks to the fuel occurred.

The available fuel inspection records consisted solely of the sipping results. An examination of selected assemblies and fuel rods was conducted by GE following Cycle 1 and has been documented in a proprietary, therefore unavailable,GE report. (NEDM-21136 Jan. 1976, Quad Cities Unit 2 End of Cycle 1 Fuel Inspection Report.)

The records of the detailed operating history were by far the most voluminous and the most difficult to obtain and to assemble in a useable form. The first set of records reviewed to obtain the power history was a monthly plot of the gross generation rate, megawatts electric, as a percentage of 850 MWe vs. time, with several points plotted each day. Many of these plots contained notes describing the reactor operation. These plots proved to be a valuable road map to the operating histories. The initial selection of the events to be analyzed was determined by reviewing these plots. Another record that helped in identifying events of interest was the compilation of notes maintained by the reactor engineer during the operating periods when the fuel preconditioning limits were exceeded.

Identifying the dates of the events was the first step in the actual power shock - failure prediction work. The second step was to review the detailed operating history to determine what power changes were made, what rates of change were used and how the changes were made. Of particular concern, and the most difficult to obtain, were the records of control rod drive movements.

All of the operating records for Quad Cities are maintained on microfilm. These include the alarm logs, the average power range monitor (APRM) rod block monitor, and the offgas activity level chart. The APRM rod block monitor charts provide a continuous record of the reactor power level and at the same time may record a significant change in local power caused by control rod drive movement. The APRMs are an average of many local power range detectors.

The rod block monitor signal is representative of the local power range monitors that surround the selected control rod. The local power range monitor (LPRM) signals are initially normalized to the APRM signal. Control rod drive motion that significantly affects the local power is then recorded as a variation in the APRM signal. It was originally thought that the record of these strip charts could be used to identify the time and, to a limited extent, the significance of control rod drive motion.

This approach, in practice, turned out to be impractical. The following actions all produce similar indications on the Average Power Range Monitor-Rod Block Monitor (APRM-RBM) strip charts:

- Selection of a control rod drive to display the output signals in the LPRM instrumentation strings adjacent to the drive selected at the operator's console.
- Selection of a control rod drive for a fully withdrawn or fully inserted control rod to permit the application of hydraulic pressure to increase cooling water flow.
- Exercising of a control rod drive (CRD) to check its functional ability.

Tracking down the real source of these signals in the alarm typewriter output proved to be a monumental task, since for the first year or so the alarm typewriter produced tens, if not hundreds, of pages of output per day.

The initial solution to the problem of identifying CRD moves was simply to use the P1 edits* from the process computer which were periodically available and to assume that the changes in fuel assembly power levels were caused by control rod moves that were conducted in accordance with the preconditioning recommendations. As will be discussed in Section 6 of this report, this initial assumption led to poor geographic resolution of the failure predictions and a second data acquisition campaign was conducted. During the second campaign additional information on CRD

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* A P1 edit is a printout from the process computer used with GE BWR's that tabulates the results of the primary system heat balance and other calculations. Of special interest to this study were the control rod drive position maps provided by this edit.

moves, particularly during 1974, was obtained from a combined review of the reactor engineer's and the reactor operations logs.

Examples of additional material obtained during the second Quad Cities campaign include:

- Two detailed reports on the operations during the rod withdrawal error of September 21, 1973 and the fuel failure event during the startup of Cycle 2, May 22, 1975.
- Core maps showing the calculated fuel bundle exposures for the beginning and end of each Cycle (these printouts by the process computer are called E-buns). These were used as a check of the ScP calculations.
- Core maps showing the calculated bundle power for various times for plant control during the first 2 cycles (these printouts by the process computer are called P-buns). These also were used as a check of the RECORD/PRESTO calculations.
- Tip traces early in each cycle with the associated heat balances.

The data obtained and their application to the analysis of the first 2 cycles of Quad Cities Unit Two operation are discussed in Section 6.

MAINE YANKEE

With a few notable exceptions, the data collection program for the first 2 cycles at Maine Yankee was a repeat of the Quad Cities program. The most important difference was the ease with which the control rod drive positions at Maine could be tracked. The control rods are operated in groups and each rod within the group is always positioned within one step (3/4") of the other rods in the group. The group positions are automatically typed on the NUCLEAR LOG every hour. Scanning these logs proved to be an effective way to determine control rod motion and since this same log also recorded the reactor power it was possible to decide at once if the rod motion was of potential significance from the power shock point of view.

Other differences at Maine were that, rather than offgas data, primary coolant activity was obtained, particularly the I-131 and I-133 concentrations. These were based on the daily sample analysis.

Post irradiation examination results consisted of sipping data and the hot cell work reported in Reference (17).

Supplemental data from Maine Yankee included primary coolant boron concentrations at various exposures throughout the first 2 cycles and power distribution calculations based on incore instruments. Both of these were used to verify the calculations of RECORD-PRESTO.

MANUFACTURING DATA

For both Quad Cities and Maine Yankee the fuel was considered to have been manufactured as designed. An indepth review of the manufacturing records could not be accomplished, because of the proprietary nature of this type of information and the difficulty in retrieving it for fuel that was manufactured several years ago. For the analysis of Quad Cities we do not believe that this limitation affected the predictions; however, the results from Maine Yankee indicate a possible manufacturing difference between fuel types of nearly identical design. These results are discussed in Section 7.

Section 5

METHODS OF ANALYSIS

THE FUEL PERFORMANCE MODEL

ScP's fuel performance model, POSHO, (1)* calculates the probability of failure of a fuel rod resulting from pellet clad interaction during a "power shock." The model employed in this study is, basically, the operational version of POSHO as it existed at the end of 1975. It was postulated originally on the basis of phenomenological data from the Halden Reactor Project and was developed and adjusted on the basis of experience from European and U. S. light water reactors. The model requires, as input, information about the nodal power shocks created in the reactor core during operation. In this study the power shock information was obtained by simulating the reactor nodal power history with Scandpower's Fuel Management System, FMS. (2). In an operating plant, the nodal power history may be obtained on line from the process computer.

The basic assumption in the POSHO model is that the reason for failure is local clad over-straining, resulting from stresses produced in the clad by thermal expansion of the UO_2 during positive local or total power changes. Stress corrosion, as a major contributing factor in the failure process is not explicitly considered by the POSHO model. In many experiments, the time of failure has been taken implicitly as the time of activity release. However, internal pressure sensors used in fuel rods at the Halden Test Reactor have detected pressure increases inside the rods when cracks first occurred, without accompanying activity releases for more than a week. Analysis of the test conditions (e.g., higher external rod pressure), and recent experiments indicate that many PCI cracks do not release activity until an intervening power cycle has occurred. Thus, a still undetermined fraction of PCI failures occurs sometime prior to detection of coolant activity rise, and, in fact, failures have been observed to occur during the power increase without associated activity release. POSHO assumes that failure occurs at the time of the power shock, or within a few hours of the shock. POSHO will underestimate the effect of a shock of low magnitude if the failure is chemically assisted.

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* References are provided in Section 9.

Thermal expansion of the UO_2 pellets is transmitted to the cladding only if the linear heat generation rate (Q) exceeds the interaction threshold level (Q_o) for which the fuel expands radially to contact the cladding. The value of Q_o is, however, a dynamic quantity which changes throughout the operating cycle, mainly dependent on the actual power Q relative to Q_o.

The tracking of Q_o with time is done by two functions which simulate the important slow dimensional change processes. These two correlations have been synthesized from in-core fuel deformation measurements at Halden.

The "fuel conditioning correlation" is used when the local power, Q, is greater than the interaction threshold power, Q_o. The clad is then in hoop tension and the fuel is subject to a radial compressive load which tends to compact the pellets. Under these circumstances, Q_o increases, approaching Q as a limit if Q is held constant. The "fuel deconditioning correlation" is used when the operating Q is less than the interaction threshold, Q_o, which means that there is a radial gap between the fuel and the clad. In this situation, the fuel is assumed to relocate outwards, reducing Q_o, which approaches Q as a limit if Q is held constant.

Figure 5-1 illustrates how the interaction threshold power density, Q_o, moves toward the local power, Q, for a section of a fuel rod where considerable power shock has occurred during two events. The power shock for any event is defined as the maximum value by which Q exceeds Q_o.

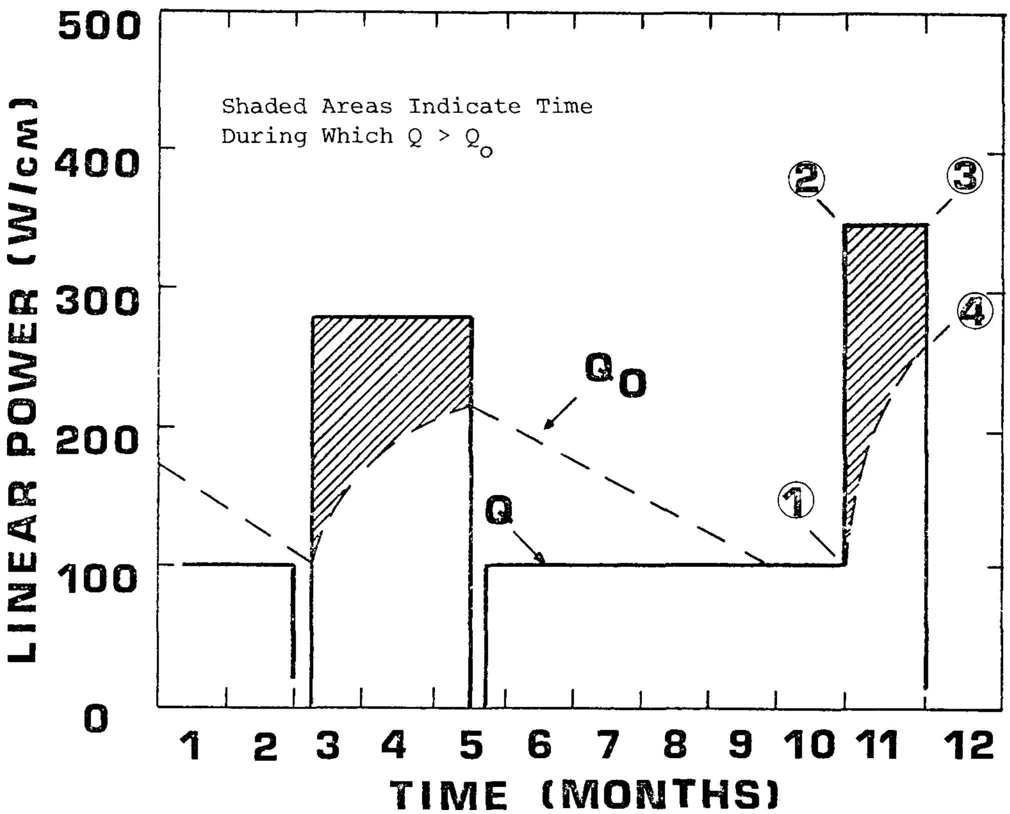

Figure 5-1. Tracking the Interaction Level, Q_o, for a Local Power Variation, Q, with Time

Figure 5-2 pictorializes the process by which failure probability is found in POSHO. The coordinates of the interaction threshold, Q_o, and the shock magnitude $\Delta Q=(Q - Q_o)$, locate the probability of failure for the fuel segment in question. The solid lines, with reference labels 1 to 4, show how the locus of failure probability moves during the event of Figure 5-1 with conditioning taking place. The vertical dashed line shows the locus of failure probability if the power rise were instantaneous.

For an instantaneous power rise, from 100 W/cm to 350 W/cm the power shock, $Q - Q_o$, would be 250 W/cm (since $Q_o = Q = 100$ W/cm initially). For this shock the probability of failure would be approximately 5×10^{-4}. For a real shock ascension the value of Q_o would increase during the time the power shock was taking place. This is represented by the heavy solid curve between points 1 and 2. Thus, the power shock is less than the total power change; in this case the power shock is 200 W/cm, and the probability of failure is reduced to approximately 5×10^{-5}. While the power, Q, is constant between points 2 and 3 at a level of 350 W/cm, the interaction threshold power, Q_o, increases to approximately 250 W/cm. Since the power shock for this event if defined as the maximum value by which the power, Q, exceeds the interaction threshold power, Q_o, the probability of failure for this event is associated with the 200 W/cm shock (i.e., 5×10^{-5}) and does not change regardless of the time spent at the increased power level. If the power ascension were slower than depicted, then the probability of failure would be less, since Q_o would be increasing over a longer period of time. Point 4 indicates the value of the interaction threshold power, Q_o, reached at the time the power, Q, was reduced to zero from point 3 (350 W/cm).

For a complex power maneuver, involving an irregular power ascension, with periods of power decrease or constant level, the POSHO model used in this study required the construction of a simplified approximation to the actual power history.

The family of probability lines is specific to a particular fuel design with a particular specific power and exposure. The probability line pattern changes dynamically during a cycle with burn-up.

The failure probability lines have, so far, been calibrated to a number of BWR operating cycles. PWR cycles have been checked, but failure data are inadequate since sipping is normally not performed.

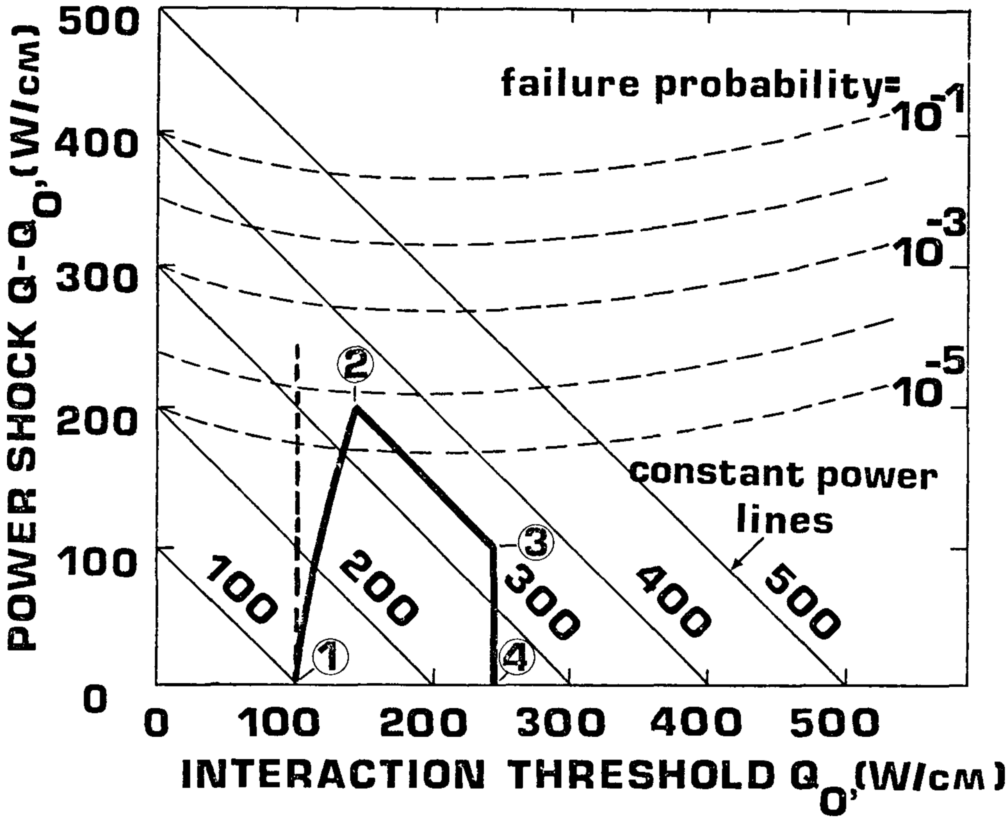

Figure 5-2. Computation of Failure Probability

FUEL DUTY CYCLE ANALYSIS

The analysis of a cycle starts with the determination of events that are believed to have produced power shocks. Examples of such events are:

- Changes in total power
- Control rod motions
- Shift in power distribution caused by increase or decrease in the concentration of xenon or other poison

A simplified total power history, which reproduces the power shocks during all of the identified events is then constructed and the nodal power histories are simulated with PRESTO.

Actual and simulated reactor power history plots are shown in Figures 6-8, 6-9, 7-5 and 7-6.

The fuel performance model, POSHO, has been integrated into Scandpower's Fuel Management System (FMS) in a subsystem called the Fuel Duty Cycle Analysis (FDCA). (See Figure 5-3.) The function of the FMS program RECORD, within this sytems, is to provide the axially integrated, relative pin power distribution in the fuel assembly, (i.e., the integrated power for each fuel pin divided by the average fuel pin power for the fuel assembly), for the control rod in/out conditions and as a function of void (BWR) and burnup. In addition, RECORD furnishes all nuclear parameters and cross-sections used by the simulator PRESTO.

RECORD treats the fuel assembly reactor physics in great detail and accounts for all lattice heterogeneities found in today's light water reactor (LWR) assembly designs, such as burnable poison (B_4C) shim rods, Gd-containing fuel rods, rodded blade cruciform control elements and rod cluster control elements.

PRESTO is a three-dimensional LWR core simulator, utilizing a nodal core subdivision for the calculation of power- and exposure distributions. The underlying neutronics model is derived directly from 2-group diffusion theory.(3) This direct method generally is considered superior to conventional nodal-coupling methods. The function of PRESTO within the FDCA system is to provide detailed, 3-dimensional power distributions as input to the power shock generation module. More detailed descriptions of the computer programs RECORD, PRESTO and FDCA, are given in Appendix A. A functional diagram of the FDCA system is provided in Figure 5-3, in which

P = nodal power

α = void coefficient

α_E = exposure weighted void coefficient

b = burnup

c = control fraction

Relative pin power distributions are combined with the nodal power distribution and the actual total core power history, as represented by the modeled power history, to produce a continuous power history for three (PWR) or four (BWR) groups of pins in each node. The pin groups are selected so that the ratio of pin powers with control rods inserted to pin powers with control rods withdrawn, within a group, is the same, within ± 5% (PWR) or ± 20% (BWR). Examples of these ratios given in Figures 6-15 and 6-16. The corresponding PCI interaction level and power shocks are calculated during each power ramp and stored in a permanent file, the $\Delta Q, Q$ file, together with the actual power history. For the quarter core simulation of

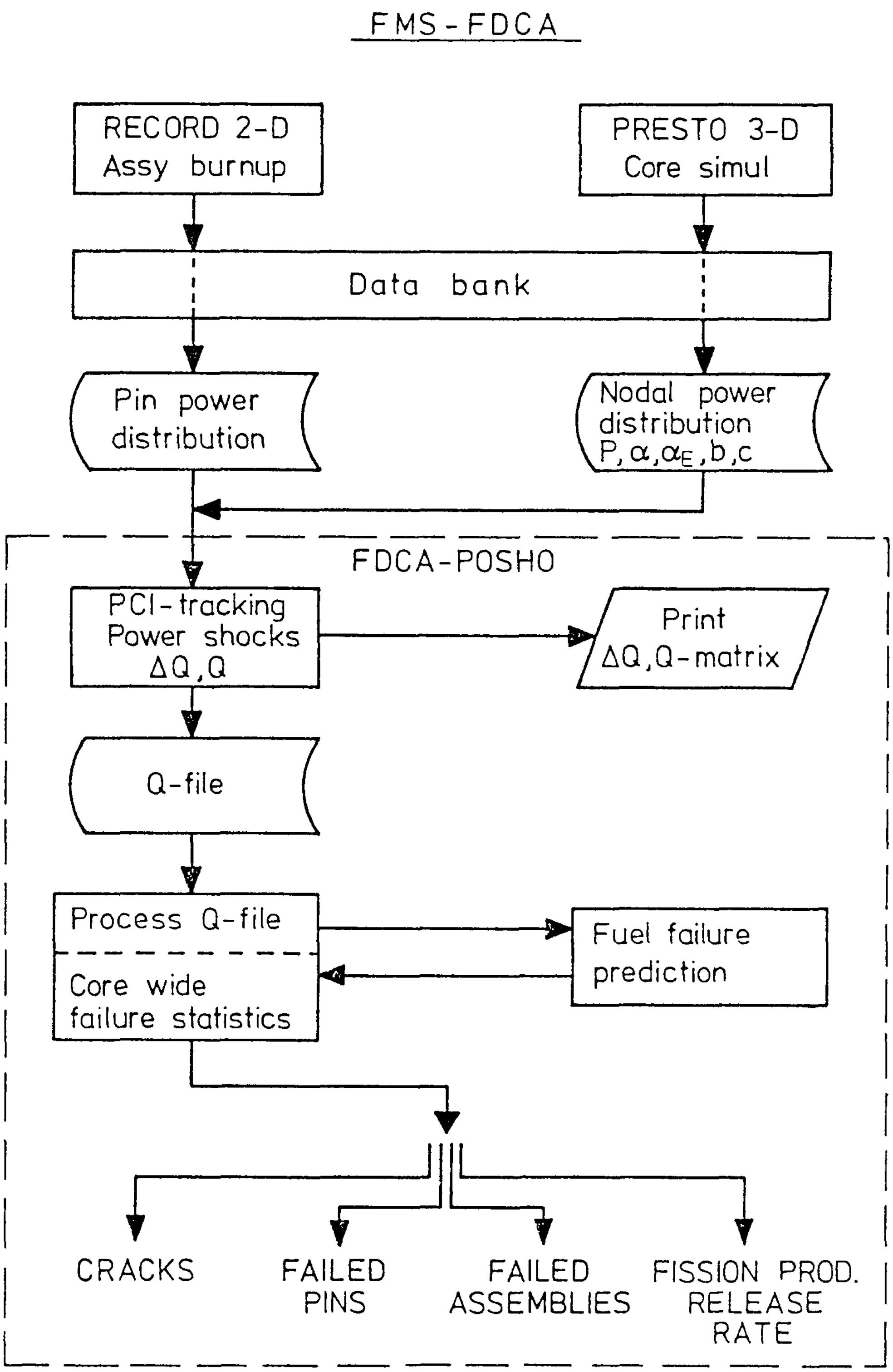

Figure 5-3. Functional Diagram of the FDCA System
Incorporating the POSHO Logic

Quad Cities Unit Two with 24 axial nodes in each assembly and 4 pin groups, this represents over 68,000 data records for each time span.

FUEL FAILURE PROBABILITY INTEGRATION

For each pin group in each node, the probability for fuel failure is evaluated by the POSHO model. The crack probabilities are integrated for each pin group in the assembly to give the expected number of failed pins and assemblies for each event and as a function of time.

Failure probabilities are integrated, using standard probability theory formulae and assuming independent failures:

The failure probability, P_k, for pin section k is given by

$$P_k = 1 - e^{-\lambda_k \cdot \Delta \ell}$$

where

λ_k = failure probability per unit length for section k.

$\Delta \ell$ = section length (node height).

The failure probability, P_i, for the whole pin is:

$$P_i = 1 - \prod_{k=1}^{KMAX} (1 - P_k)$$

where KMAX is the number of axial nodes in the assembly.

The failure probability, P_m, for pin group m, containing N_m pins is:

$$P_m = 1 - (1 - P_i)^{N_m}$$

where it is assumed that the pin probabilities, P_i, are equal, for the N_m pins making up the group.

where it is assumed that the pin probabilities, P_i, are equal, for the N_m pins making up the group.

The assembly failure probability, P_ℓ, is

$$P_\ell = 1 - \prod_{m=1}^{NG} (1 - P_m)$$

with NG pin groups in the assembly.

The expected number of failed assemblies, NA_c, is:

$$NA_c = \sum_{\ell=1}^{LMAX} P_\ell$$

where LMAX is the total number of assemblies analyzed.

The expected number of cracks, NC_k, in section k of a pin is:

$$NC_k = \lambda_k . \Delta\ell$$

For pin group m, this is:

$$NC_m = N_m \cdot \sum_{k=1}^{KMAX} (\lambda_k . \Delta\ell)$$

The expected number of cracks, NC_ℓ, in the assembly is:

$$NC_\ell = \sum_{m=1}^{NG} NC_m$$

and for the whole core:

$$NC_c = \sum_{\ell=1}^{LMAX} NC_\ell$$

Correspondingly, the number of failed pins may be calculated from the pin failure probabilities.

Accumulation of probabilities over a number of events, j, is obtained by:

$$P_{acc} = 1 - \prod_{j=1}^{JMAX} (1-P_j)$$

where P_{acc} is the probability accumulated over JMAX independent events.

FISSION PRODUCT RELEASE ALGORITHM

Activity Release. Transport Mechanisms and Interpretation of Measurement.

The fission products are generated within the UO_2 matrix in the pellets. The high temperatures and large temperature gradients within a pellet tend to move the most volatile nuclides to the grain boundaries and along the grain boundaries to the gas space surrounding the pellets by diffusion processes. Some fission products will react chemically with the surroundings, while others will solidify and accumulate in the colder regions of the fuel. Transport and accumulation of fission products in various regions of the fuel rods are affected by the decay of unstable nuclides, forming other isotopes, which in turn may be unstable and undergo decay. In the gas-space of a leak-tight rod, the accumulation of a radioactive isotope will approach an asymptotic value if the rod is operated at a steady-state power condition. The concentration of each isotope is then determined by the production and the decay rates of the specific isotope. When leakage occurs, this steady state condition is changed, as the hold-up time in the rod is now finite. Different isotopes are affected differently by a leak, because of the different decay times. The isotopic spectrum which is released from a leaking rod is a function of the leak rate and is therefore dependent on the size of the crack. It ranges from an equilibrium spectrum when leakage is small to a diffusion type spectrum when the leakage is large (hold-up time depends, then, mainly on the diffusion velocity within the grains of the UO_2 pellets). When a new rod starts leaking, and the leak is large, the isotopic spectrum will change in a transient manner towards an equilibrium spectrum. If UO_2 material is released to the coolant, the fission products will be released directly to the water, and a recoil spectrum results.

Activity Release Measurements in Power Reactors

The behavior of fission product gases in both sound and failed fuel during reactor operation has received considerable attention. In its state-of-the-art report on the role of fission gas release in reactor licensing, the U. S. Nuclear Regulatory commission (NRC) cites some 90 references.(4) Much of the work has been concerned with the release of both the stable and the radioactive gases from the oxide fuel into the pellet-clad gap. The primary interest of these studies was in the effects of these gases on the fuel temperature and the fuel rod pressure and on how they affect the fuel performance during normal, transient and accident conditions.

The study of the release of the radioactive fission gases from failed fuel has received renewed attention in recent years in response to regulatory requirements to minimize the release of these gases into the environment. The application of the results of these studies is documented in the various licensing submittals made by the reactor manufacturers.(5,6,7,8) The pressurized water reactor suppliers all use some variation of the escape rate coefficient method that relies on the earlier work at the Westinghouse Bettis Laboratory. In a very general way this method provides the probability that an isotope produced during fission will escape to the coolant. Additional information on fuel performance in PWRs sometimes is obtained from studying the ratio of the concentrations of I^{131} to I^{133} in the primary coolant. The boiling water reactor supplier, General Electric (GE), has studied the releases of radioactive fission gases from failed fuel as they have been determined from samples taken at the steam jet air ejector (SJAE) for a number of years.(9) In a BWR it is convenient to use the measurements of isotopes of the noble gases, xenon and krypton, as they do not react chemically and have a low solubility in water. Measurements of these isotopes at the SJAE are preferred since the transport time from a leaking rod to the measuring device is significantly lower than the decay time, except for short-lived isotopes. GE has developed a so-called "diffusion" mixture of isotopes, which is between the equilibrium (or long delay time) concentration and the recoil (or no delay time) concentration. The release rate of this mixture is based on diffusion theory and is a function of each isotope's radioactive decay constant (λ) raised to the one half power ($\lambda^{0.5}$). This is between lambda to the first power for a recoil mixture and lambda to the 0 power for the equilibrium mixture. GE (10) now uses an empirically determined equation with $\lambda^{0.4}$.

In a PWR, it is more convenient to use the isotopes of iodine with a longer decay time and a high solubility in the water. The hold-up time in the primary coolant is relatively large due to the small purification flow, and has to be accounted for when interpreting the results. During transient operations, iodine activity release resulting from water flushing large amounts of iodine out into the coolant may dominate the activity level completely. It is therefore necessary to analyze results from steady-state operation to avoid large errors.

Thus, the methods in use today (10), (11), (12) do not consider the type of fuel failures, their location, or the local operating conditions in the vicinity of the failures. Operator-developed correlations, based on fission product activity measurements and coupled with core duty cycle analyses would tell when, where, and what type of failure had taken place and would thus help to reduce failures and to locate failures during refueling outages. The purpose of the fission product release algorithm study for this project was to make a start toward the development of methods that would provide the operator with a better understanding of fuel performance from activity measurements made during reactor operation.

The first phase of the work was to conduct a survey of the literature much of which has been cited above. The other noteworthy references are listed in Section 10, Bibliography. To obtain additional background on the state-of-the-art, discussions were held with each of the reactor suppliers: Babcock and Wilcox, Combustion Engineering, General Electric and Westinghouse. Reference was made mostly to testimony at the ALAP hearing and to licensing submittals.

B&W, however, discussed two computer codes that they have under development to track fission product inventory and to calculate release rates, one for steady state and one for dynamic conditions. A brief review of these codes, however, indicate that they would not be directly useful in this phase of the work. In addition to their code work, B&W has done some development work on on-line primary coolant activity monitors. (13)

Xenon Isotope Concentration in BWR Offgas

During the course of the search for methods that might provide operators with a better understanding of fuel performance during operation, ScP learned of a technique used at KRB's Gundremmingen Boiling Water Reactor (a 237 MWe dual cycle Dresden 1-type). (14) At this facility the offgas is analyzed isotopically on a daily basis. Records are maintained for the total offgas activity (μCi/sec) and isotopic analysis is done for the 7 principal isotopes. The data are further reduced in an effort to determine the type of fuel failure.

In addition to the general methods for determining the distribution between recoil mixtures and equilibrium mixtures, the ratio of Xe^{133} to Xe^{135} is calculated for each sample. For a recoil mixture this ratio would be less than 0.1 and for an equilibrium mixture the theoretical ratio is about 4 at the SJAE. The expected value given by GE for a large BWR operating with failed fuel is about 0.4 (15), and at KRB the ratio was noted to vary from 0.3 to 0.5 with spikes over 1.0.

The most important use made of the xenon ratio at KRB is in determining which areas of the core contain the failed fuel. Prior to a refueling outage a core map is marked to indicate the control rods whose motion apparently caused large spikes in the xenon ratio. The fuel cells containing these rods are then the prime targets for the in-core sipping program. If the sipping of these cells yields the expected number of fuel failures, the remainder of the core is not sipped. The Grundremmingen experience with this limited sipping has indicated that it is as effective as the earlier total core sipping program. This program is very cost-effective in that the sipping program is on the critical path for refueling in over 75% of the outages.

In order to learn more of the usefulness of this xenon ratio, the offgas data from the first 2 cycles at Quad Cities were reviewed and this ratio was calculated for all samples for which the isotopic analysis was available. The results of these calculations are shown in Table 5-1. A review of this table shows the normal value to be in the range of 0.4 to 0.5 with periodic spikes over 1.0. Since the initial application of POSHO at Quad Cities indicated that not all the events from 1974 had been accounted for, this table was used in the search for power shock events during the year 1974. The correlation of the events found with xenon ratios over (or near) 1.0 is shown in the 1974 section of Table 5-1.

It appears that power shocks cause a marked increase in the ratio of these xenon isotopes, which indicates a shift in the offgas mixture from the recoil towards the equilibrium mixture. It is not known if this shift is caused from the initiation of new failures, the opening of existing cladding cracks, or simply a change in the temperature of the fuel in the vicinity of existing cracks. Many other questions remain on how the ratio changes for a given shock (e.g., when does the change start; how long does it last; and what peak values are reached?). Answers to these questions would require acquisition of data on xenon isotope concentrations as a function of time and power shock history.

The Activity Release Algorithm

In the first attempt at developing a fission product algorithm, emphasis was placed on the importance of operating conditions in the fuel in the region of the defect. The general assumptions used in developing the algorithm are as follows:

- The fission product inventory available for release was considered constant (proportional to the local power level), i.e., depletion caused by the leakage through a defect was neglected.
- The fission product release rate from the fuel to the gap was considered to be proportional to a power of the local linear heat generation rate, i.e., to Q^m, where m is a constant determined empirically.
- The leakage from the cladding was considered to be directly proportional to the calculated local probability for pin failure.
- The calculated results were normalized to plant operating data.

At steady state power operation, the local fission product inventory of a fuel pin is given by:

$$N = K_1 Q$$

where N = nuclides per cm of fuel pin

K_1 = a constant

Q = local pin power - w/cm.

The local pin leakage, given a local failure probability, P, is given by:

$$L = \nu_N \cdot N \cdot P$$

where L = local pin leakage rate - nuclides/sec.

ν_N = the escape rate coefficient for the N^{th} nuclide - sec^{-1}.

P = the local pin failure probability

Although ν_N is a complex function of diffusion, axial transport resistance and crack leakage properties, it is assumed to be proportional to an exponential function of the local power, i.e.,

$$\nu_N = K_2 Q^m$$

where K_2 = a constant

Substituting for ν_N and N, the local leakage may be expressed as:

$$L = KPQ^{m+1}$$

The total leakage from one fuel pin becomes:

$$L_{pin} = K\int_{z=o}^{z_{max}} PQ^{m+1} dz$$

where z is the axial height.

The leakage rate for the entire core is then the sum of the leakages from all pins:

$$L_{core} = \Sigma_{pins} L_{pin} \qquad (1)$$

where L_{core} = the total core leakage - μCi/sec

The value of m used in this study is 1.5, so that the total release rate is considered to be proportional to the local power raised to the 2.5 power. This value of m was determined from a study of the primary coolant activity at Maine Yankee during a power reduction in December 1974, and also was used for the Quad Cities analysis.

Equation 1 may be used directly in the calculation of BWR release rates; however, in a PWR where the primary coolant iodine concentration is being calculated it is necessary to consider the accumulation, decay and removal of each isotope whose concentration is being considered. (16) This was done using the following equation:

$$N_c = \frac{L_{core}}{V(\lambda_N + \beta e_N)}$$

where

N_c = fission product nuclide concentration in the coolant (nuclides/m^3)

V = Reactor Coolant System (RCS) volume (m^3)

β = purification rate in the primary coolant purification system (m^3/sec/RCS m^3)

e_N = purification efficiency for the N_{th} radionuclide

L_{core} = as defined by Eq. (1)

λ_n = radioactive decay constant (sec^{-1})

TABLE 5-1

QUAD CITIES UNIT TWO
OFFGAS DATA

CYCLE 1 - 1973

Ratio of Xe^{133} to Xe^{135}

Date	Ratio	Date	Ratio	Date	Ratio	Date	Ratio	Date	Ratio	Date	Ratio
Jan 2	0.60	May 7	0.79	Jul 19	1.33	Sep 23	0.30	Nov 4	0.89	Dec 11	0.77
11	0.53	9	0.86	20	1.20	23	1.10	5	0.64	11	0.63
16	1.80	19	4.05	25	1.28	24	0.56	6	0.63	12	0.64
17	0.96	21	0.54	26	1.57	25	0.39	7	0.78	13	0.50
18	1.11	23	0.93	Aug 6	1.84	26	1.39	8	0.67	14	0.43
22	1.67	24	0.89	7	0.51	27	0.56	9	0.68	16	0.47
24	1.12	25	1.19	6	2.13	28	0.51	10	0.63	17	0.49
29	67.00	26	0.87	8	0.61	29	0.44	11	0.54	18	0.46
30	1.53	27	1.14	9	0.71	30	2.70	12	0.60	19	0.49
Feb 3	3.15	27	1.26	13	1.21	Oct 1	0.51	13	0.66	20	0.48
5	6.77	28	0.54	14	0.94	2	0.35	14	1.49	21	0.56
7	1.60	30	0.81	16	0.82	3	0.49	15	1.12	22	0.57
10	6.05	Jun 1	0.72	20	0.92	4	0.40	15	0.31	23	1.10
12	15.50	2	0.95	21	0.90	5	0.49	16	0.42	24	1.38
14	8.12	3	0.68	22	0.91	6	0.61	17	0.94	25	0.87
16	22.20	4	1.42	23	1.02	7	0.50	18	2.99	26	0.63
20	8.46	6	0.84	24	1.21	8	0.48	18	0.74	27	0.74
23	0.64	8	1.78	26	1.13	9	0.45	19	0.40	28	0.65
27	1.85	8	0.78	27	5.13	10	1.53	19	0.38	29	0.75
28	1.97	11	0.92	28	1.17	11	0.46	20	0.70	30	0.59
Mar 1	1.66	11	0.93	29	1.02	12	0.51	21	0.36	31	0.62
2	1.17	14	1.21	30	0.98	13	0.89	22	0.38		
3	1.49	14	1.60	31	1.17	14	0.44	23	0.40		
9	1.88	18	9.85	Sep 2	1.54	15	0.46	23	0.38		
12	3.77	19	1.06	3	1.87	16	0.55	24	1.51		
13	3.35	20	0.84	4	1.18	23	1.09	25	0.46		
15	1.32	20	1.60	5	1.43	23	0.44	26	0.46		
19	3.52	22	1.21	6	1.67	24	0.27	26	0.44		
20	1.99	25	2.33	7	1.79	24	0.21	27	0.56		
23	0.81	26	0.98	10	3.96	25	0.24	28	0.51		
26	2.38	27	1.49	11	1.59	26	0.91	29	0.48		
28	2.08	29	1.23	12	0.93	27	0.27	30	0.53		
30	4.03	Jul 2	2.58	13	0.73	27	0.68	Dec 1	0.58		
Apr 2	3.27	3	1.43	14	0.65	28	0.34	2	0.52		
2	4.76	5	1.31	15	0.65	28	0.40	3	0.65		
5	3.27	9	1.51	16	0.57	29	2.78	4	0.70		
6	0.81	9	1.69	18	0.81	30	0.47	5	0.59		
8	1.40	10	1.69	19	0.83	30	0.49	6	0.64		
10	1.86	11	1.70	20	0.67	Nov 1	0.53	7	0.63		
12	4.94	12	1.66	21	0.67	2	0.73	9	0.66		
13	4.36	14	1.66	22	1.11	3	0.88	10	0.55		
May 1	5.95	16	1.80	22	2.04	4	1.05				
3	10.00	17	1.35								
5	0.35	18	1.11								

Table 5-1 (Cont.)

QUAD CITIES UNIT TWO
OFFGAS DATA

CYCLE 1 - 1974
Ratio of Xe^{133} to Xe^{135}

Date	Ratio		Date	Ratio	
Jan 1	0.61		Feb 22	0.48	
2	0.65		23	0.63	
4	0.62		24	0.52	
7	4.16	PC. Envelope Exceeded	25	0.50	
8	0.55		26	0.85	
9	0.36		27	0.48	
10	0.44		28	0.42	
11	0.50		Mar 1	0.42	
12	0.59		2	0.92	
13	0.51		3	0.48	
14	0.48		4	0.52	
15	0.45		5	0.84	
16	0.55		6	0.45	
17	0.55		7	0.44	
18	0.62		8	0.39	
19	0.68		9	0.47	
20	0.55		10		
21	0.58		11	0.47	
22	0.65		12	0.47	
23	0.55		13	0.45	
24	0.12		14	0.45	
28	7.58	PC. Envelope Exceeded	15	0.43	
29	0.94		16	0.47	
30	0.35		17	0.45	
31	0.33		18	0.41	
Feb 1	0.35		19	0.32	
2	0.37		20	0.44	
3	0.77		21	0.41	
4	0.49		22	0.42	
5	0.81		Apr 1	1.22	Startup
6	0.48		2	0.37	
7	0.63		3	0.38	
8	0.61		4	2.33	CRD Moves
9	1.00	CRD Moves for Shaping	5	0.32	
10	1.23	" " " "	6	0.43	
11	0.88		7	0.73	
12	0.72		8	0.90	
13	0.67		9	0.75	
14	0.41		10	0.71	
15	0.47		11	0.75	
16	0.45		12	0.70	
17	0.45		15	2.13	CRD Moves
18	0.45		16	0.22	
19	0.56		17	0.32	
20	0.43		18	0.58	
21	0.45		19	0.56	

Table 5-1 (Cont.)

QUAD CITIES UNIT TWO
OFFGAS DATA

CYCLE 1 - 1974
Ratio of Xe^{133} to Xe^{135}

Date	Ratio		Date	Ratio	
Apr 20	0.39		Jun 6	0.35	
21	0.49		6	0.36	
22	0.55		6	0.37	
23	0.64		7	0.38	
24	0.52		7	0.34	
25	0.49		7	0.61	
26	0.50		7	0.47	
27	0.41		8	0.43	
28	0.43		8	0.62	
29	0.51		9	0.41	
30	0.46		17	14.63	Startup
May 1	0.44		18	0.89	
2	0.48		18	0.34	
3	0.46		25	0.20	
4	0.85	CRD Moves - Low Power	26	0.19	
7	0.63		27	0.24	
7	0.71		28	0.20	
8	0.63		29	0.26	
9	0.49		29	0.33	
9	0.55		30	0.32	
10	0.56		Jul 3	2.38	Startup
11	0.65		4	0.26	
12	0.57		5	0.27	
13	0.66		5	0.44	
14	0.74		6	0.54	
15	1.11	Power increase w/flow	7	0.42	
16	0.49		8	0.38	
17	0.55		9	0.43	
17	0.30		10	0.54	
19	0.38		11	0.41	
20	0.51		12	0.40	
21	0.40		13	0.73	MSIV Surveillance
22	0.61		14	0.48	
23	0.43		15	0.45	
26	2.40	Fast return to Power	16	0.81	CRD Motion Low Power
26	0.70		17	0.17	
26	0.32		18	0.69	
27	0.35		18	0.50	
27	0.48		19	0.44	
28	0.21		20	0.40	
29	0.27		20	0.64	
30	0.27		21	0.49	
31	0.29		22	0.50	
Jun 1	0.33		23	0.56	
3	0.50		24	0.47	
3	0.55		25	0.51	
4	0.33		26	0.55	
5	0.45		27	0.60	
5	0.52		28	0.51	

Table 5-1 (Cont.)

QUAD CITIES UNIT TWO
OFFGAS DATA

CYCLE 1 - 1974

Ratio of Xe^{133} to Xe^{135}

Date	Ratio		Date	Ratio	
Jul 29	0.50		Sep 27	0.30	
30	0.47		28	0.19	
31	0.53		29	0.19	
Aug 1	0.54		29	0.37	
2	0.51		30	0.27	
3	1.27	Power reduction & return	30	0.40	
4	0.70	w/flow control	Oct 1	0.17	
5	0.41		2	0.25	
5	0.37		2	0.48	
6	0.39		3	0.23	
7	0.50		4	0.28	
8	0.49		4	0.36	
9	0.60		5	0.31	
10	1.29	Same as Aug. 3	6	0.38	
11	0.64		7	0.42	
12	0.84		8	0.42	
12	0.54		9	0.41	
13	0.43		10	0.43	
14	0.60		11	0.44	
14	0.60		12	0.37	
15	0.51		13	0.34	
15	0.47		16	0.50	
15	0.45		16	0.32	
16	0.44		16	0.31	
17	0.39		17	0.38	
17	1.07	Same as Aug. 3	18	1.42	Scram recovery
18	0.52		18	0.37	
19	0.49		18	0.49	
19	0.54		19	0.37	
20	0.38		19	0.35	
21	0.54		20	0.26	
22	0.50		21	0.44	
23	0.47		22	0.32	
24	0.47		23	0.37	
25	0.57		24	0.50	
26	0.49		24	0.33	
27	0.85		24	0.41	
28	0.50		25	0.44	
29	0.51		26	0.50	
30	0.47		27	0.48	
Sep 8	0.76	Startup	27	0.43	
8	0.33		28	0.50	
9	0.39		28	0.43	
10	0.30		29	0.46	
11	0.38		30	0.58	
12	0.41		30	0.46	
13	0.59		31	0.49	
26	0.53		Nov 1	0.62	

Table 5-1 (Cont.)

QUAD CITIES UNIT TWO
OFFGAS DATA

CYCLE 1 - 1974

Ratio of Xe^{133} to Xe^{135}

Date	Ratio		Date	Ratio	
Nov 2	0.57		Dec 15	0.48	
3	0.52		16	0.41	
4	0.40		17	0.55	
5	0.39		18	0.42	
6	0.41		19	0.43	
7	0.42		20	0.45	
8	0.38		21	0.45	
9	0.36				
10	0.39				
11	0.34				Abbreviations:
12	0.36				
13	0.33				PC. - Preconditioning
14	0.46				CRD - Control Rod Drive
15	0.32				MSIV - Main Steam Isolation Valve
15	0.44				
16	1.27	Pwr.reduction & ret.w/CRD			
17	1.30	CRD withdrawal for shaping			
18	0.45				
19	0.41				
20	0.40				
21	0.41				
22	0.50				
23	0.29				
24	0.38				
25	0.52				
26	0.44				
26	0.33				
27	0.94	Small power increase			
28	0.55				
28	0.41				
29	0.44				
30	0.38				
Dec 1	0.44				
2	0.40				
3	0.36				
4	0.37				
5	0.42				
5	0.42				
6	0.41				
7	0.40				
8	0.51				
9	0.44				
10	0.77	Power decrease & return			
11	0.36				
12	0.45				
13	0.44				
14	0.48				

Table 5-1 (Cont.)

QUAD CITIES UNIT TWO
OFFGAS DATA

CYCLE 2 - 1975

Ratio of Xe^{133} to Xe^{135}

Date	Ratio		Date	Ratio		Date	Ratio		Date	Ratio	
Apr 26	0.06		Jun 12	0.46		Jul 8	1.08		Aug 9	0.41	
28	0.00		13	0.37		9	0.42		10	186.00	*
30	0.07		13	18.65	*	10			10	3.98	
May 1	0.07		14	16.13	*	10	0.48		11	0.45	
2	0.09		15	10.48	*	11	155.50	*	11	153.33	*
3	0.05		15	13.84	*	11	0.38		12	0.42	
4	0.08		16	25.63	*	12	310.00	*	13	0.47	
4	0.14		16	0.32		12	233.00	*	13	232.50	*
6	0.43		18	51.67	*	12	0.41		13	0.48	
7	0.17		19	0.46		13	312.33	*	13	185.60	*
8	0.13		19	0.39		13	0.40		27	2.65	
9	0.16		19	57.86	*	14	0.44		28	0.28	
10	0.04		20	0.39		14	185.60	*	28	0.29	
11	0.18		21	147.67	*	15	154.67	*	29	0.22	
12	0.18		21	0.47		16	232.25	*	29	157.80	*
14	0.28		22	0.41		16	0.53		30	94.38	*
15	0.21		22	64.08	*	17	131.86	*	30	0.29	
19	0.28		22	0.47		17	0.40		31	0.23	
22	1.03		24	0.44		18	185.20	*	31	195.50	*
22	1.37		24	219.25	*	18	0.42		Sep 6	197.40	*
22	2.30		25	0.49		19	0.71		6	1.55	
22	2.13		25	220.00	*	19	5.33		7	0.24	
22	1.86		26	0.47		20	300.00	*	8	75.13	*
22	2.23		26	141.67	*	20	1.03		8	0.23	
23	1.31		26	147.00	*	25	2.04		8	0.31	
26	0.78		27	0.42		25			9	79.13	*
26	0.74		28	0.85		25			9	0.28	
26	0.65		28	218.00	*	25	0.47		10	0.32	
27	0.57		29	0.41		27	0.43		10	75.50	*
26	0.60		29	71.17	*	28	0.49		11	78.88	*
29	0.84		30	101.25	*	29	0.37		11	0.48	
30	1.05		30	0.07		29	0.40		12	0.38	
30	0.99		Jul 1	0.48		29	219.25	*	12	53.00	*
Jun 1	162.67	*	1	118.00	*	30	0.41		13	0.40	
1	0.52		2	122.14	*	30	103.75	*	13	31.79	*
2	97.90	*	2	0.45		Aug 1	0.37		13	57.22	*
2	0.49		3	0.44		1	132.50	*	14	0.45	
3	60.63	*	3	125.00	*	2	0.63		15	0.68	
5	9.57	*	4	75.00	*	2	101.63	*	15	38.00	*
5	0.52		5	0.45		3	0.73		16	0.63	
6	6.37	*	5	304.67	*	3	478.00	*	16	20.48	*
7	5.15	*	6	0.55		4	0.72		17	23.63	*
8	3.66	*	6	230.50	*	4	994.00	*	17	0.40	
9	3.46	*	7	184.00	*	5	0.51		18	13.29	*
9	0.25		8	0.48		6	492.00	*	18	0.52	
10	2.85	*	6	231.00	*	6	0.47		19	0.54	
11	6.00	*	7	0.54		7	0.36		19	19.88	*
11	0.81		8	232.25	*	8	0.39		20	0.52	
12	28.64	*	8	0.46		8	188.60	*	21	0.62	
						9	206.25	*	22	13.77	*

TABLE 5-1 (Cont.)

QUAD CITIES UNIT TWO
OFFGAS DATA

CYCLE 2 - 1975

Ratio of Xe^{133} to Xe^{135}

Date	Ratio		Date	Ratio
Sep 22	0.98			
23	28.52	*		
23	0.49			
23	65.50	*		
24	0.50			
24	72.62	*		
25	64.91	*		
25	0.58			
26	91.88	*		
26	0.49			
27	82.78	*		
27	0.50			
28	0.50			
28	77.40	*		
29	0.50			
29	113.57	*		
30	0.52			
30	63.67	*		
Oct 1	120.71	*		
1	0.50			
2	144.50	*		
3	147.17	*		
3	0.58			
2	144.17	*		
2	0.50			

* These ratios are based on an isotopic analysis of a sample taken at the stack. These samples are much older than the SJAE samples and are probably not of any significance in the study of fuel performance.

Section 6

QUAD CITIES UNIT TWO

DESIGN AND PERFORMANCE DATA

Commercial operation at Quad Cities 2 started in May 1972. Since May 1973 the plant has been operated using GE's Preconditioning Interim Operating Management Recommendations.

On 21 September 1973, the operator inadvertently withdrew a single rod which should have remained in the core at 33% insertion. The rod was out for 5 hours before it was reinserted. This event is the most important single event during Cycle 1, from a fuel model testing viewpoint. At the end of Cycle 1, all assemblies were sipped ex-core and 74 leaking assemblies were identified and removed.

Early in Cycle 2, the unit was started up after a short (∿24 hour) outage. The plan was to come up at 100 MWe/hr to 400 MWe, and then at 50 MWe/hr to 700 MWe, or to the precondition envelope, whichever came first. Power was then to be increased at 4 MWe/hr. The rod pattern selected was not appropriate for the core conditions at the time of restart, because it resulted in exceeding the preconditioning envelope. This was not detected at the time, because power distributions provided by the process computer were not available. Commonwealth Edison's early estimate of the fuel damage from this event was that as many as 100 assemblies may have failed. The unit remained derated to about 80% power for the remainder of the cycle. The cycle ended in the fall of 1975, with a fuel maintenance outage. All assemblies were sipped in-core and a total of 94 failed assemblies identified.

Descriptions of the design and performance data used in this study are given below. The initial core loading consisted of 724 fuel assemblies with the same average enrichment of 2.12% and 7x7 fuel rod arrays. Two rods per assembly contained Gd_2O_3 over the full fuel length. One additional part length gadolinium rod was used in 312 of the core I assemblies. Dished fuel pellets were used in a total of 461 assemblies.

The cycle 2 reload fuel consisted of 144 8x8 assemblies with an average enrichment of 2.50%. The subdivision of the core loading into fuel types for modelling purposes is described in this section under the heading Core Model.

Fuel and Assembly Descriptions

- Geometric design data for fuel pellets, rods and fuel assemblies for the initial 7x7 fuel and the reload 8x8 fuel are given in Table 6-1.
- Corresponding material composition data are given in Table 6-2.
- Arrangements of fuel rod types for each fuel assembly type are shown in Figures 6-1, 6-2 and 6-3.

Control Rod Design

- A cross-section of the control rod blade is shown in Figure 6-4, together with a list of the most important dimensions.

Core Design

- Table 6-3 identifies the total number of assemblies, control rods, heat transfer areas, etc.
- The core layout and fuel type identification for Cycle 1 is given in Figure 6-5.
- The core layout and fuel type identification for Cycle 2 is given in Figure 6-6.
- Figure 6-7 is a drawing showing the elevation of relevant core components.

Nominal Operating Conditions

- Nominal values of core thermal power, coolant flow rate, coolant sub-cooling, etc., and operating limits are given in Table 6-4.

Operating History

- The gross thermal power production history through Cycles 1 and 2 is shown in Figures 6-8 and 6-9, together with the simplified power history, as modeled for this study.
- Data sets, giving core average burnup, heat balance data and control rod patterns, describing the steady-state core condition on a number of selected dates through Cycles 1 and 2, are given in Appendix B.
- Plots of the offgas activity recordings through Cycles 1 and 2 were prepared. Examples of such plots are given in Figures 6-10 and 6-11.
- Figure 6-12 shows the locations of the assemblies identified as leakers by the sipping analysis for Cycle 1.
- Figure 6-13 shows the locations of the assemblies identified as leakers by the sipping analysis for Cycle 2.

For more detailed descriptions and drawings of fuel assemblies and core components, the reader is referred to Reference (15).

DUTY CYCLE SIMULATION

Nuclear Data Bank Generation

RECORD calculations were performed to generate a complete nuclear data bank, consisting of 2-group macroscopic cross-sections, diffusion coefficients and pin-power distributions, all as functions of burnup up to 35000 MWD/MTU and for three voids, 0, 40% and 70%.

Separate RECORD calculations were done to determine special model coefficients, as used in PRESTO for the effects of control rods, Doppler equilibrium xenon and transient xenon.

Examples of k_∞ versus burnup for each of the three fuel assembly arrangements, shown in Figures 6-1, 6-2 and 6-3, are given in Figure 6-14. These curves are for 40% void.

The subdivisions of the fuel assemblies into pin-groups for the initial 7 x 7 fuel with 3 Gd pins, and for the reload 8 x 8 fuel, are shown in Figure 6-15 and 6-16. Four pin groups were used, both for the 7 x 7 and for the 8 x 8 fuel.

Polynomial expansions of the radial pin-group power distributions were generated as functions of burnup and void for the rod-out condition, by means of the FMS polynomial generator, POLGEN. Ratios of rodded-to-unrodded pin-group power were calculated for the fresh fuel at 40% void, as shown in Figures 6-15 and 6-16, and were taken to be independent of burnup and void.

Core Model

A 1/4-core model, consisting of 181 assemblies, each subdivided into 24 axial nodes, was set up for simulation with PRESTO. Four different fuel types were defined for the initial core in order to distinguish between the dished and undished fuel for the 2 and 3 Gd pin fuel assemblies, respectively. The fuel type definition and layout is shown in Figure 6-5. The axial Gd distribution for pin type Gd2 was explicitly taken into account, however, the 3-inch Gd-free end sections of pin type Gd1 were not accounted for; i.e., the Gd-content was assumed to extend over the entire length for these pin types.

The second core was modeled with 5 fuel types, where Fuel Type 5 represents the reload fuel, as shown in Figure 6-6. Fuel shuffling within the simulated core quadrant, as well as asymmetric shuffling across the quadrant symmetry lines, was accounted for. Shuffling from "out-of-quadrant" core positions was simulated by means of the fuel history "reflection" option, as built into the PRESTO simulator.

The reactor operating condition at a given point in time was characterized by an "operating condition data set," consisting of a control rod pattern and some key heat balance data, as shown in Appendix B. Such data sets were constructed from the station process computer output data files. In-core instrumentation recordings were used to check the adequacy of the calculated power distributions. The PRESTO simulator includes a routine for comparison with the in-core fission chamber detectors.

Comparisons with a set of traveling in-core probe (TIP) curves were made near the beginning of Cycle 1 (BOC-1) and again near the BOC-2. Results of the measured and calculated TIP traces are shown in Figures 6-17 and 6-18. During the Cycle 1 simulation, the calculated power distributions were also checked versus the evaluated LPRM recordings from the process computer. An example of this is shown in Figure 6-19.

DUTY CYCLE SIMULATION

The operating history was subdivided into 16 timesteps through Cycle 1, and 6 timesteps through Cycle 2, for the purpose of representing the fuel exposure distribution. A timestep is characterized by an operating period with approximately the same rod pattern. During periods of changing core reactivity, the rods may have been moved a few notches during the timestep. The power distribution is therefore calculated at beginning and end of the step. The calculated k_{eff} versus accumulated core burnup through Cycles 1 and 2 is shown in Figure 6-20.

Core power distributions were generated and stored in the data bank for a total of 46 points in time.

The fuel duty cycle was then simulated by combining these core power distributions with the pin group power distributions, as obtained by RECORD and the approximated core power histories given in Figures 6-8 and 6-9, using the FDCA-1 program (see Section 4). The fuel duty cycle is described by a sequence of events that may have produced local or global power shocks.

The events were selected on the basis of information on rod motions and reactor power changes obtained from the operator log books, process computer outputs, etc. The following were considered as events:

- Reactor operation for fuel preconditioning (precon. ramps),
- Total power increase, where the nature and magnitude of the increase may have produced power shocks,
- Startup with a power distribution different from that existing prior to shutdown,
- Rod movements at power, and
- Power distribution shifts due to xenon transients.

Rod movements at power were often difficult to describe properly, due to incomplete data on rod positions as a function of time. For the period prior to May 1973, it was assumed that the net rod movements, from one operating condition data set to the next had occurred at power. For the period after May 1973, when the Preconditioning Interim Operating Management Recommendations provided by GE (PCIOMR) became effective, it was assumed that rods were moved at reduced power.

The rod withdrawal event, which occurred in September 1973, was subject to an hour-by-hour simulation to account for the xenon transient during this event. The fuel assemblies surrounding the withdrawn rod experienced power shocks that exceeded, by a factor of 2, any other power shock identified during Cycle 1. The power distributions in one of these assemblies, before and after the event, are shown in Figure 6-21.

The start-up event on May 22, 1975, early in Cycle 2, is believed to be responsible for the bulk of the failures which occurred in that cycle. This event was modeled in detail, as shown in Figure 6-22, using the xenon dynamics mode of simulation. The reactor was taken up to 700 MWe, or about 85% of full power, with an adverse bottom peaked power distribution. (See Appendix C) The bottom peak was especially pronounced in Core Quadrant No. 2, where almost no shallow rods were used to reduce the power peaking. This quadrant was analyzed separately in order to account approximately for the asymmetries relative to Quadrant No. 4.

A list of the events analyzed is given in Table 6-5 for Cycle 1, and in Table 6-6 for Cycle 2. Each event is characterized by the reactor power level before and after the event, the ramp time and type of event, such as total power increase, rod movement, etc. Detailed power distribution and power shock data for selected events are given in Appendix C in the form of copies of the computer output listings.

An important part of the duty cycle analysis is to track the interaction power level, Q_o, for each nodal pin group as a function of time. An example of this is shown in Figure 6-23, which illustrates the development of Q_o for the pin group adjacent to the control rod (Group 4) in Node No. 6 of Assembly 903 through Cycle 1, together with the power, Q, and the effective power shocks ΔQ for this pin group. Assembly 903 was located next to the control rod involved in the rod withdrawal event of September 1973.

Corresponding values of Q_o, Q and ΔQ for all 17,376 pin group nodes in the core quadrant considered and for all events analyzed, were generated and stored on a permanent file, called the Q-file.

FUEL PERFORMANCE ANALYSES

Failure Prediction

The Q-file, containing the duty cycle history, was processed by the FDCA2/POSHO program to obtain the final failure predictions, failure accumulation and the predicted offgas activity level versus time.

Tables 6-7 and 6-8 list the maximum power shocks, the expected number of cracks, failed pins and failed assemblies per event for the events in Cycles 1 and 2, respectively. Since some events included rather complex power histories, the simulated power history was chosen to produce the maximum shock that would be judged to occur for the event. Figures 6-24 and 6-25 show the accumulated assembly failure probability distributions in the simulated core quadrant, at the end of Cycle 1 (EOC1) and at the end of Cycle 2 (EOC2), respectively.

For Cycle 1, the failed fuel assemblies from this quadrant, as well as from symmetric positions in the other quadrants, are identified for comparison. A breakdown of the failed assemblies into fuel types and core quadrants is shown in Table 6-9 for Cycle 1, and Table 6-10 for Cycle 2. The actual number of failed assemblies (sipping) for Quadrant 4 serves as a basis for comparison with the prediction. For Cycle 1, a prediction for the full core was made by extrapolation from the 1/4-core calculation.

For Cycle 2, the large spread in the observed number of failed assemblies per core quadrant indicates that asymmetric conditions existed. The failure rate in Quadrant No. 2 was more than a factor of 2 higher than in any of the three other quadrants. Hence, an attempt was made to treat this quadrant separately, as described above under Duty Cycle Simulation. The results of this calculation are shown in Figure 6-26. The extrapolation to a full core prediction was made by multiplying the failures predicted for Quadrant 4 by three and adding the prediction of Quadrant 2.

The ability of the fuel performance model to predict when the failures occur is illustrated in Figure 6-27 for Cycle 1, and in Figure 6-28 for Cycle 2. The calculated offgas activity level versus time is here compared to the measured offgas represented by the sum of six isotopes. Also shown is the accumulated prediction of number of failed pins. The core-wide distribution of the expected number of failed pins at EOC-1 is shown in Figure 6-29.

Fission Product Release Calculations

The offgas activity level from the sum of the six isotopes: Xe^{133}, Xe^{135}, Xe^{138}, Kr^{85m}, Kr^{87} and Kr^{88}, at the steam jet air ejector (SJAE) has been calculated with the model described in Section 4.

Based on data from Maine Yankee and Quad Cities, a common escape rate coefficient (m) of 1.5 was selected to describe the power dependency of the activity release rate in both reactors.

The calculated activity level is normalized to the activity level following the rod withdrawal event of 21 September 1973. This event is well defined and gives a major increase in the number of pins failed. A total of 37 pins is predicted to have failed as a result of this and prior events. Agreement with the average offgas level during the subsequent period of steady-state operation, Step 16, is obtained using a K value in Eq.5 in § 4 of $1.3 \cdot 10^{-5}$. This corresponds to an average release rate of 1400 μCi/sec per pin at 100% power.

The same offgas model was employed in the calculations for Cycle 2. The results for Cycle 1 are shown in Figure 6-27 and the results for Cycle 2 are shown in Figure 6-28. During the first year and a half of Cycle 1, the predicted offgas activity is in relatively good agreement with the measured activity and both correlate quantitatively with the calculation of the number of failed fuel pins. Late in Cycle 1 and in Cycle 2, there is very wide scatter in the measured activity levels, which is typical for this kind of measurement in a BWR, and only gross correspondence with the calculated number of failed fuel pins.

EVALUATION OF RESULTS

The Power History Data Base

The procedure of the fuel duty cycle analysis for developing a detailed power history data base relies on the ability of the analytic models of FMS to simulate core performance and the accuracy with which the operating history is simulated.

The ability of FMS to adequately account for the complex effects of power level, flow, control rod distribution and exposure history on core reactivity is clearly demonstrated by a nearly constant k_{eff} value throughout the two cycles analyzed. The calculated k_{eff} as a function of burnup is shown in Figure 6-20. This provides an integral confirmation of the adequacy of the nuclear data base produced with RECORD.

Comparisons between measured and calculated in-core signals show that the PRESTO simulator adequately reproduces the power distributions in the core throughout the two cycles. The axial power distributions for ten locations in one quadrant of the core, as measured and calculated at the beginning of Cycles 1 and 2, are shown in Figures 6-17 and 6-18. The standard deviation between the calculated and measured local TIP signal was 7-8% for the comparisons shown in these figures. The results achieved in the core simulation phase of this work may deserve more attention, but in the context of this work it is sufficient to note these results as a verification of the adequacy of the simulator in reproducing the actual power distributions in the core.

Simulation of the operating history for a fuel duty cycle analysis requires, basically, a normal core-follow analysis to describe the state of the core, as a function of time, and additional simulation of any operating event which could produce gross or local fuel shocks during the operating period.

In the approximation of the operating history shown in Figures 6-8 and 6-9, gross core power maneuvers are considered to be well represented by the steady-state power distributions at beginning and end of each timestep shown. During these time steps, however, single notch withdrawals of rods at power have occurred. Considerable effort was expended in the identification of these events. Some uncertainty is also associated with transient power distributions which may have occurred during some of the startup events. This was particularly true for Cycle 1, where the operating history was quite complex. Cycle 2 represents a much simpler cycle where the significant events are considered to be well modeled.

In spite of these difficulties, the power history data base, developed for Quad Cities 2, is considered to be the most comprehensive and accurate data base available for further development of empirical failure models like POSHO.

Power Shock Calculations

The power history data base was used to model the selected power shock events using the FDCA1 program. Thus the effective interaction level history and effective power shock at each event were determined. Table 6-7 gives the maximum power shock at each of these events. The uncertainty associated with the modeling of power shocks is determined essentially by the uncertainty associated with the calculation of a change in local power, which is considered to be smaller than the uncertainty (7-8%) in the power distribution itself. The introduction of the operating recommendations to maintain power distributions within preconditioning values has not prevented power shocks from occurring frequently. This result is not surprising, in view of the complexity of the three-dimensional power distribution in a BWR. More guidance, particularly in the predictive capability of the process computer, is required to enable the operator to do a better job in controlling the power shape.

Failure Analysis

The gross core failure predictions by FDCA-POSHO, Tables 6-9 and 6-10, show an over-prediction of the failure rate in Cycle 1, and under-prediction in Cycle 2. For the two cycles together, the total predicted number of failures is in close agreement with observations.

As described in Section 4, POSHO has been adjusted to failure data from a number of operating plants, with an approximate representation of the power shocks developed in these plants. The detailed representations of power histories for Quad Cities have shown that the fuel in an operating BWR is subject to a much greater number of small power shocks than previously assumed. It was thus expected that the general level of failure prediction would be high in Cycle 1, because of the cumulative effect of combining the probabilities of failure of a large number of small shocks. In Cycle 2, this is compensated for by neglecting the influence of previous damage to old fuel, as discussed below.

The sipping procedure is not perfect; perhaps the ex-core sipping procedure is about 80% effective. Using this value, we find the "corrected" failure rates in Cycle 1 to be 74/0.80=92, which would imply that 18 failed assemblies remained in the core for Cycle 2. The corrected failure rate in Cycle 2 is thus $(\frac{94}{0.80} - 18)$ = 100 failed assemblies, for a corrected total of 192 failed assemblies for the two cycles. Compared with these values, the over-prediction in Cycle 1 is reduced to about 10%. There is still, however, an under-prediction in Cycle 2 of approximately 40%. This is attributed primarily to the fact that FDCA-POSHO does not account for the cumulative effects of previous power shocks in the fuel's resistance to PCI failures. This effect also has been observed in other applications of POSHO, and may be a result of previous deteriorization of the fuel cladding and previous development of incipient cracks.

Looking at the failures for the different assembly types, it is seen that in Cycle 1, Type 2 assemblies have experienced more than twice as many failures on a percentage basis, as the nearly identical assemblies of Type 4. Type 2 assemblies are predicted to have about the same failure rate as Type 4. There is no known design difference that can explain the better behavior of the Type 4 assemblies relative to the Type 2 assemblies. It does not seem reasonable that only the power shocks for Type 2 assemblies have been under-estimated. This difference may, therefore, come from design or production variables not accounted for in the model, and perhaps is an illustration of the possible differences which can be expected between two different batches produced according to the same specifications. In Cycle 2, this effect has disappeared and/or the damage incurred during Cycle 1 results in a fuel state giving nearly the same failure rates for the two fuel types.

Assemblies of Type 3 experienced few failures in Cycle 1, but have the highest failure rate in Cycle 2. One possible reason for this is that they were originally loaded in the outer ring, and many of them were shifted into the center of the core for operation in Cycle 2.

Table 6-10 shows the division of the failures in Cycle 2 among the four quadrants. It can be seen that there is a large difference in the number of observed failures for the four quadrants. In Quadrant 2, 41 assemblies failed, but in Quadrant 3, only 15 failed. The power simulation and failure prediction was done for Quadrant 4, which is reasonably representative for the power history in Quadrants 1, 3 and 4. A separate analysis was done for Quadrant 2 for the startup event, where a higher peak was developed in this quadrant, due to an asymmetric control rod distribution.

This improved the prediction for this quadrant, but it remains below the observed values.

Comparisons between the accumulated number of predicted failed pins and the offgas activity versus time show a positive correlation for both Cycles 1 and 2. In most cases, the increase in offgas coincides with the time of new failures predicted, or occurs shortly thereafter. The activity level calculated from the predicted number of failed pins is generally in agreement with the observed offgas level as a function of time. There are, however, indications of an over-prediction of the number of fuel failures occurring during the preconditioning maneuvers, based on the observed offgas activity level changes. A greater effect of preconditioning may be required in POSHO for slow ramp rates.

Turning, finally, to the model's ability to predict where the fuel failures occur in the core, it is observed that the picture is somewhat complex. During Cycle 1, almost every fuel assembly in the central core experienced a considerable number of shocks and the general failure probability level is high. About half of the failed assemblies were found in only one of four symmetric positions. There still remain, however, some locations in the core where symmetric assemblies have failed in all four quadrants. A better geographic resolution of the location of the predicted failed assemblies would have been desirable.

The geographic resolution and correlation with observations is excellent in Cycle 2 in both of the quadrants analyzed. This is due to the fact that in this cycle, the bulk of the failures were caused by a few large shocks, rather than many small ones.

The same conclusions may be drawn from the "Error Plots" of Figures 6-30 and 6-31. Here, the observed failure rate for assemblies with predicted failure probabilities within the same interval is presented. Making allowance for the general low level of the failure prediction in Cycle 2, there is a strong correlation between prediction and observation for both quadrants analyzed.

The predictions may be used to eliminate "safe" assemblies from the time consuming sipping procedure. It is, however, important to note that in Cycle 1, where most of the assemblies received significant power shocks, only about 100 assemblies out of 720 had predicted failure probabilities of less than 5%. In Cycle 2, however, nearly half of the assemblies could have been eliminated from the sipping process on the basis of predictions of the probability of failure of 5% or less.

As already discussed, the failure probabilities in Cycle 2 have been calculated assuming no effect of previous shocks. The predicted failure probabilities in Quadrant 4 at EOC-2 are presented in Figure 6-32 together with the failure probabilities of the respective assemblies from EOC-1. It is observed that in no case has fuel failed in Cycle 2 without receiving power shocks resulting in a failure probability

A correlation which would account for the cumulative effect of power shocks would provide a multiplicative correction to the failure probability functions. The damage correction, itself, would be an integral function of power shock history. This would be in agreement with physical understanding of the failure process and work is underway along these lines.

Fission Product Release

The predicted offgas activity level shows a reasonable agreement during the first part of Cycle 1, while there is an under-prediction toward the end of Cycle 1. There also is a consistent under-prediction of the activity level throughout Cycle 2. The model over-predicted the number of failed assemblies in Cycle 1. There is, however, no information about the true number of pins failed during the cycle, and the predictions may be low. Because of the wide scatter in the measured values, numerical comparison is difficult. The under-prediction is roughly 50% at the end of Cycle 1 and the predicted level is very roughly 25% of the measured level during Cycle 2. A damage correction in the POSHO model probably will result in more pins failed per assembly. The core was subjected to large local power shocks early in the cycle, and there also were several rod motions at power later in the cycle affecting fuel which may have had earlier incipient failures.

The present activity release model does not include a burnup dependency. There are, however, several reasons that call for an increased activity release rate at high burnup. The fuel pellets become more cracked. Accumulation of fission products in the UO_2 matrix and in the fuel to cladding gap will tend to reduce specific heat conductivity, and gap conductance. (18) The center temperature therefore increases with burnup. These factors will tend to increase the escape rate coefficient and give higher release rates for the same failure rate, in the absence of other factors (e.g., crud) that would decrease the release rates.

Fuel pins which have failed early in the cycle will tend to form new cracks, because subsequent hydriding will weaken the cladding. This also may increase the release rate.

In Cycle 2, the predictions underestimate the number of failed assemblies and consequently, also the number of failed pins and the resultant offgas activity level. The algorithm does not consider the activity release from fuel material released to the primary coolant system from prior failures. This omission contributes to the under-prediction of that activity. For an under-prediction of failures of 35%, the actual number of failures would be expected to increase the predicted activity level by approximately 50%. If the escape rate is increasing with burnup, the large discrepancy between the measured and the predicted offgas activity level may be reduced substantially.

Table 6-1

Geometric Design data for the initial 7 x 7 fuel and the reload 8 x 8 fuel. (All dimensions in cm.) Quad Cities Unit Two

DESCRIPTION	INITIAL FUEL	RELOAD- 1 FUEL
Fuel pellet radius	.6185/.6198*	.5283
Fuel pellet height	2.134	1.067
Dish radius	.4826	
Pellet - clad gap	.0152/.0139	.0114
Clad, inner radius	.6337	.5398
Clad, thickness	.0813	.0863
Clad, outer radius	.7150	.6261
Fuel rod pitch	1.8745	1.6256
Fuel rod array	7 x 7	8 x 8
Outer side of fuel assembly box	13.8125	13.8125
Fuel assembly box thickness	.2032	.2032
Fuel assembly pitch	15.24	15.24
Thickness of narrow water gap	.4750	.4750
Thickness of wide water gap	.9525	.9525

*Dished/Undished

TABLE 6-2

FUEL AND ASSEMBLY MATERIAL COMPOSITION FOR INITIAL AND RELOAD FUEL. QUAD CITIES UNIT TWO

DESCRIPTION	INITIAL FUEL	RELOAD-1 FUEL
Weight of UO_2 pr. assembly (kg)	218.1/223.0*	208.383 (Dished)
Fuel density (UO_2) (g/cm^3)	10.4	10.4
Fuel enrichment, wt % U235 in U	2.12	2.50
Uranium in fuel (wt %)	88.15	88.15
Clad type	Zircaloy 2	Zircaloy 2
Clad density (g/cm^3)	6.55	6.55
Material in fuel assembly box	Zircaloy 4	
Density of fuel box material (g/cm^3)	6.55	
Structural material	7 spacers (Zr-4 with inconel springs)	

*Dished/undished

Table 6-3

Quad Cities Unit Two
Core Design Data

Total number of fuel assemblies	724
Number of throttled periphery channels	84
Total heat transfer area (m^2) - Core 1	5819.446
Heat transfer area per assembly (m^2), 7x7	8.038
Heat transfer area per assembly (m^2), 8x8	9.065
Total number of control rods	177
Control rod pitch (cm)	30.48
Core height (active fuel) (cm)	365.76

Table 6-4

Nominal Operating Conditions

Core thermal power (MW)	2511
System pressure (psia)	1029
Steam production (kg/sec)	1230.4
Total coolant flow rate (kg/sec)	12348
Core inlet subcooling (wsec/kg)	50233

Table 6-5
Description of Events - Cycle 1 QUAD CITIES UNIT TWO

Event No.	Date of Event	Power Level (%) Before Ramp	After Ramp	Ramp* Time Hrs.	Description of Event
0	BOC(1972)	0.	90.2	300.	Startup. Rod patt. A1
	13/1-1973	0.	90.	120.	Rod swap to B1
	23/1	0.	96.	6.	Rods moved on power
	26/1	0.	95.6	120.	Startup
	20/3	95.6	94.7	2.	Rods moved on power
5	7/5	0.	97.6	190.	Rod swap to A2 (low axial)
	7/5	97.6	98.8	0.	Rods moved on power
		98.8	94.7	96.	
	17/5	0.	93.9	76.8	Precond (high axial)
	25/6	93.9	93.9	6.	Rods moved on power
10	5/8	0.	93.4	24.	Fast startup
	8/9	0.	89.6	192.	Rod swap to B2 (low sweep)
	17/9	0.	89.4	216.	Precond. (flat sweep)
	21/9	89.4	89.4	0.	Rod withdrawal error
	21/9	89.4	89.4	4.	Max power peak (Xe-transient)
15		89.4	89.4	72.	
		0.	90.3	12.	Fast startup & rod movement
	27/12	53.	89.8	115.2	Precond. ramp (B2b)
	8/1-1974	0.	93.6	24.	Fast startup
	12/2	47.5	74.9	76.8	Precond. ramp (B2b)
20	12/2	74.9	79.4	0.	Rod withdrawal Array 30
	13/2	79.4	92.4	28.8	Increase to full power
		92.4	92.4	6.0	Rods moved on power
	5/4	0.	94.5	96.	Rod sequencing to A1
	6/4	31.	65.	2.4	One circ pump operating
25	15/4	0.	98.3	19.2	Fast power increase
	15/4	98.3	98.3	0.	Rods moved on power
		98.3	98.3	12.	Rods moved on power
	7/5	49.	97.5	120.	Precond. A1
	27/5	0.	91.	48.	Offgas HI HI

(Continued next page)

TABLE 6-5 (continued)

Event No.	Date of Event	Power Level (%) Before Ramp	After Ramp	Ramp* Time Hrs.	Description of Event
30	1/6	0.	82.	12.	Fast startup
	24/6	0.	89.6	240.	Rod swap to B1
	15/7	49.	91.	48.	Precond. B1 (low axial)
	12/8	47.5	78.5	48.	Precond. B1
	13/8	78.5	78.5	0.	Rods moved on power
35		72.	90.	24.	
	13/10	0.	82.5	240.	Power increase limits
	6/11	62.5	84.	60.	Power limited by offgas
	22/11	0.	75.7	72.	Rod swap to A2
	22/11	48.	75.	48.	Precond. ramp
40		28.	67.6	48.	One circ. pump oper.

*An effective ramp time is used which is determined mainly from the ramp rate at high power

TABLE 6.6 DESCRIPTION OF EVENTS - CYCLE 2

QUAD CITIES UNIT TWO

Event No.	Date of Event	Power Level (%) Before Ramp	Power Level (%) After Ramp	Ramp Time Hrs.
1	5/5-1975	0.	80.	120.
2	16/5	50.	90.	144.
3	22/5	0.	85.	10.5
4	22/5	85.	85.	0.
5	22/5	85.	60.	0.
6	12/6	60.	75.	10.
7	21/7	0.	73.	96.
8	14/8	0.	80.	216.

1	Startup CY2 Preconditioning ramp.
2	Preconditioning ramp.
3	Fast startup with high power peaking and asymmetric rod geometry
4	Rod movements at power
5	Low power operation due to offgas limitations
6	Power increase
7	Startup
8	Startup

Table 6.7

QUAD CITIES UNIT TWO

Maximum power shocks and predicted failures per event - Cycle 1

EVENT NO.	ΔQmax (W/cm)	FAILURE PREDICTION		
		Q4 Assys.	Pins	Cracks
0	0.	0.	0.	0.
	125.	0.1	0.1	0.1
	175.	0.1	0.1	0.1
	125.	0.3	0.3	0.3
	225.	0.7	0.8	0.8
5	125.	0.5	0.5	0.5
	175.	0.	0.	0.
	75.	0.	0.	0.
	100.	0.2	0.2	0.2
	100.	0.	0.	0.
10	125.	0.2	0.2	0.2
	125.	0.6	0.6	0.6
	75.	0.2	0.2	0.2
	350.*	3.0	20.5	24.6
	50.	0.	0.	0.
15	50.	0.	0.	0.
	100.	0.5	0.5	0.5
	100.	0.9	0.9	0.9
	100.	0.9	0.9	0.9
	100.	0.3	0.3	0.3
20	125.	0.1	0.1	0.1
	50.	0.3	0.3	0.3
	50.	0.	0.	0.
	125.	3.1	3.2	3.2
	25.	0.	0.	0.
25	100.	5.4	5.5	5.5
	125.	0.2	0.2	0.2
	125.	0.4	0.4	0.4
	100.	1.3	1.3	1.3
	50.	0.	0.	0.

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* Assymetric Quadrant 3 and 4 only

TABLE 6-7 (CONTINUED)

(Continued)

EVENT NO.	ΔQmax (W/cm)	FAILURE PREDICTION Q4 Assys.	Pins	Cracks
30	50.	0.	0.	0.
	100.	1.2	1.2	1.2
	50.	0.3	0.3	0.3
	125.	3.2	3.4	3.4
	125.	0.1	0.1	0.1
35	100.	1.8	1.8	1.8
	75.	0.3	0.3	0.3
	50.	0.5	0.5	0.5
	125.	1.9	2.0	2.0
	75.	0.6	0.6	0.6
40	25.	0.	0.	0.

Table 6.8

QUAD CITIES UNIT TWO

Maximum power shocks and predicted failures per event - Cycle 2

EVENT NO.	ΔQmax (W/cm)		FAILURE PREDICTION					
			Assemblies		Pins		Cracks	
	Q2	Q4	Q2	Q4	Q2	Q4	Q2	Q4
1	125.	125.	2.3	2.3	2.4	2.4	2.4	2.4
2	100.	100.	1.7	1.7	1.7	1.7	1.7	1.7
3	175.	175.	12.4	8.1	15.2	9.3	15.2	9.3
4	50.	200.	0.	1.2	0.	2.5	0.	2.6
5	0.	0.	0.	0.	0.	0.	0.	0.
6	75.	75.	0.6	0.6	0.6	0.6	0.6	0.6
7	100.	100.	0.7	0.7	0.7	0.7	0.7	0.7
8	50.	50.	0.4	0.4	0.4	0.4	0.4	0.4

Table 6.9

Actual and Predicted Failures by Assembly Type and Core Quadrant - Cycle 1, QUAD CITIES UNIT TWO

ASSEMBLY TYPE		1	2	3	4	SUM
NO. IN CORE		120	192	140	272	724
NO. OF FAILED ASSYS IN QUADR.	1	2	9	2	7	20
	2	3	12	2	2	19
	3	2	8	1	6	17
	4	2	9	1	6	18
PREDICTION QUADR.	4	3.4	9.7	1.0	12.5	26.6
FAILED TOTAL CORE		9	38	6	21	74
PER CENT		7.5	19.8	4.3	7.7	10.2
PRED. TOTAL CORE		13.6	35.5	4.0	47.9	101.0
PER CENT		11.3	18.5	2.9	17.6	14.0

TABLE 6.10

ACTUAL AND PREDICTED FAILURES BY ASSEMBLY TYPE AND CORE QUADRANT - CYCLE 2, QUAD CITIES UNIT TWO

ASSEMBLY TYPE		1	2	3	4	5	SUM
NO. IN CORE		88	144	132	216	144	724
NO. OF FAILED	1	2	5	6	5	0	18
ASSYS IN	2	4	6	16	15	0	41
	3	4	3	5	3	0	15
QUADR.	4	1	4	8	7	0	20
PREDICTION	2	2.5	3.7	4.9	6.3	0.0	17.4
QUADR.	4	1.7	2.9	4.5	5.2	0.0	14.3
FAILED TOTAL CORE		11	18	35	30	0	94
PER CENT		12.5	12.5	26.5	13.9	0.0	13.0
PRED. TOTAL CORE		7.6	12.4	18.4	21.9	0.0	60.3
PER CENT		8.6	8.6	13.9	10.1	0.0	8.3

TABLE 6.11

ACTUAL AND PREDICTED NUMBER OF FAILED ASSEMBLIES, SUM OF CYCLES 1 AND 2, QUAD CITIES UNIT TWO

ASSEMBLY TYPE	1	2	3	4	5	SUM
NO. OF ASSEMBLIES	120	192	140	272	144	868
NO. OF FAILED ASSYS. PER CENT	20 16.7	56 29.2	41 29.3	51 18.8	0 0.0	168 19.4
PREDICTION (NO.) PER CENT	21.2 17.7	47.9 24.9	22.4 16.0	69.8 25.7	0.0 0.0	161.3 18.6

WIDE-WIDE CORNER

1	1	2	2	2	2	1
1	2	2	3	3	3	2
2	2	3	3	3	3	3
2	3	3	3	3	Gd 1	3
2	3	3	3	3	3	3
2	3	3	Gd 1	3	3	3
1	2	3	3	3	3	2

Rod Type	Enrichment w/o U-235	w/o Gd_2O_3	Number of Rods
1	1.20	-	5
2	1.70	-	14
3	2.47	-	28
Gd 1	2.47	3.0	2

Fig. 6.1 Fuel assembly arrangement for initial 7 x 7 fuel with 2 Gd-pins.

WIDE-WIDE CORNER

1	1	2	2	2	2	1
1	2	2	3	3	3	2
2	2	Gd 2	3	3	3	3
2	3	3	3	3	Gd 1	3
2	3	3	3	3	3	3
2	3	3	Gd 1	3	3	3
1	2	3	3	3	3	2

Rod Type	Enrichment w/o U-235	w/o Gd_2O_3	Number of Rods
1	1.20	-	5
2	1.70	-	14
3	2.47	-	27
Gd 1	2.47	3.0	2
Gd 2	2.47	0.5	1

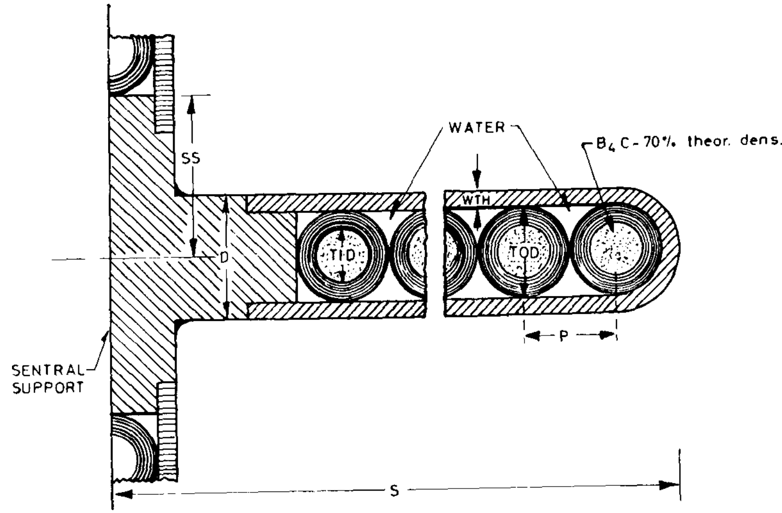
Fig. 6.2 Fuel assembly arrangement for initial 7 x 7 fuel with 3 Gd-pins.

WIDE-WIDE CORNER

1	2	3	3	3	3	3	2
2	3	4	4	4	4	4	4
3	4	GD 3	4	4	4	GD 3	4
3	4	4	4	4	4	4	4
3	4	4	4	H_2O	4	4	4
3	4	4	4	4	4	4	4
3	4	GD 3	4	4	4	GD 3	4
2	3	4	4	4	4	4	3

ROD TYPE	ENRICHMENT w/o U-235	w/o GD_2O_3	NUMBER OF RODS
1	1.40	-	1
2	1.80	-	4
3	2.06	-	14
4	2.73	-	40
GD 3	2.73	1.5	4
H_2O	-	-	1

FIG. 6.3 FUEL ASSEMBLY ARRANGEMENT FOR THE RELOAD 8 X 8 FUEL.



S	CONTROL BLADE HALF SPAN	12.3825 CM
D	CONTROL BLADE FULL THICKNESS	.7925
SS	CENTER PIECE HALF SPAN	1.9837
WTH	SHEATH THICKNESS	.1854
TID	TUBE INNER RADIUS	.175
TOD	TUBE OUTER RADIUS	.2385
P	PITCH OF ABSORBER RODS WITHIN BLADE	.4884
	NUMBER OF TUBES PER BLADE	84

FIG. 6.4 CROSS-SECTION OF CONTROL ROD BLADE (DIMENSIONS IN CM.).

QUAD CITIES UNIT TWO

2	4	2	4	2	4	2	4	2	4	2	4	2	4	3
4	1	4	1	4	1	4	1	4	1	4	2	1	3	3
2	4	2	4	2	4	2	4	2	4	2	4	2	3	3
4	1	4	1	4	1	4	1	4	1	4	2	4	1	3
2	4	2	4	2	4	2	4	2	4	2	4	1	3	3
4	1	4	1	4	1	4	1	4	2	4	2	3	3	
2	4	2	4	2	4	2	4	2	4	2	3	3		
4	1	4	1	4	1	4	1	4	2	4	1	3		
2	4	2	4	2	4	2	4	2	4	2	3	3		
4	1	4	1	4	2	4	2	4	1	3	3			
2	4	2	4	2	4	2	4	2	3	3				
4	2	4	2	4	1	3	1	3	3					
2	4	2	4	1	3	3	3	3						
3	1	3	1	3	3									
3	3	3	3	3										

FUEL TYPE IDENTIFICATION

Bundle type	No of Gd pins	Pellets	Average enrichment
1	3	UNDISHED	2.12
2	3	DISHED	2.12
3	2	UNDISHED	2.12
4	2	DISHED	2.12

FIG. 6.5 CORE LAYOUT AND FUEL TYPES IN QUAD CITIES 2 - CYCLE 1.

4	2	3	2	3	4	3	4	3	2	3	4	3	4	3
2	5	2	5	4	5	4	5	4	5	3	5	3	5	4
3	2	2	1	2	4	2	2	2	1	4	3	4	1	4
4	5	1	5	1	5	2	5	3	5	4	5	3	5	2
3	4	2	4	4	4	1	3	4	4	1	4	1	2	2
2	5	1	5	4	5	1	5	1	5	4	5	3	2	
3	4	2	4	4	4	1	3	2	3	4	2	4		
1	5	4	5	3	5	3	5	2	5	1	5	2		
3	4	4	3	4	4	1	1	3	4	2	2	4		
4	5	2	5	4	5	3	5	4	3	4	4			
3	3	4	4	2	2	2	1	2	1	4				
1	5	3	5	4	5	4	5	4	2					
3	3	4	3	1	3	1	4	4						
2	5	1	5	4	2									
2	4	2	4	2										

FUEL TYPE IDENTIFICATION

Bundle type	No of Gd pins	Pellets	Average enrichment
1	3	UNDISHED	2.12
2	3	DISHED	2.12
3	2	UNDISHED	2.12
4	2	DISHED	2.12
5	4	CHAMFERED	2.50

FIG. 6.6 CORE LAYOUT AND FUEL TYPES IN QUAD CITIES 2 - CYCLE 2.

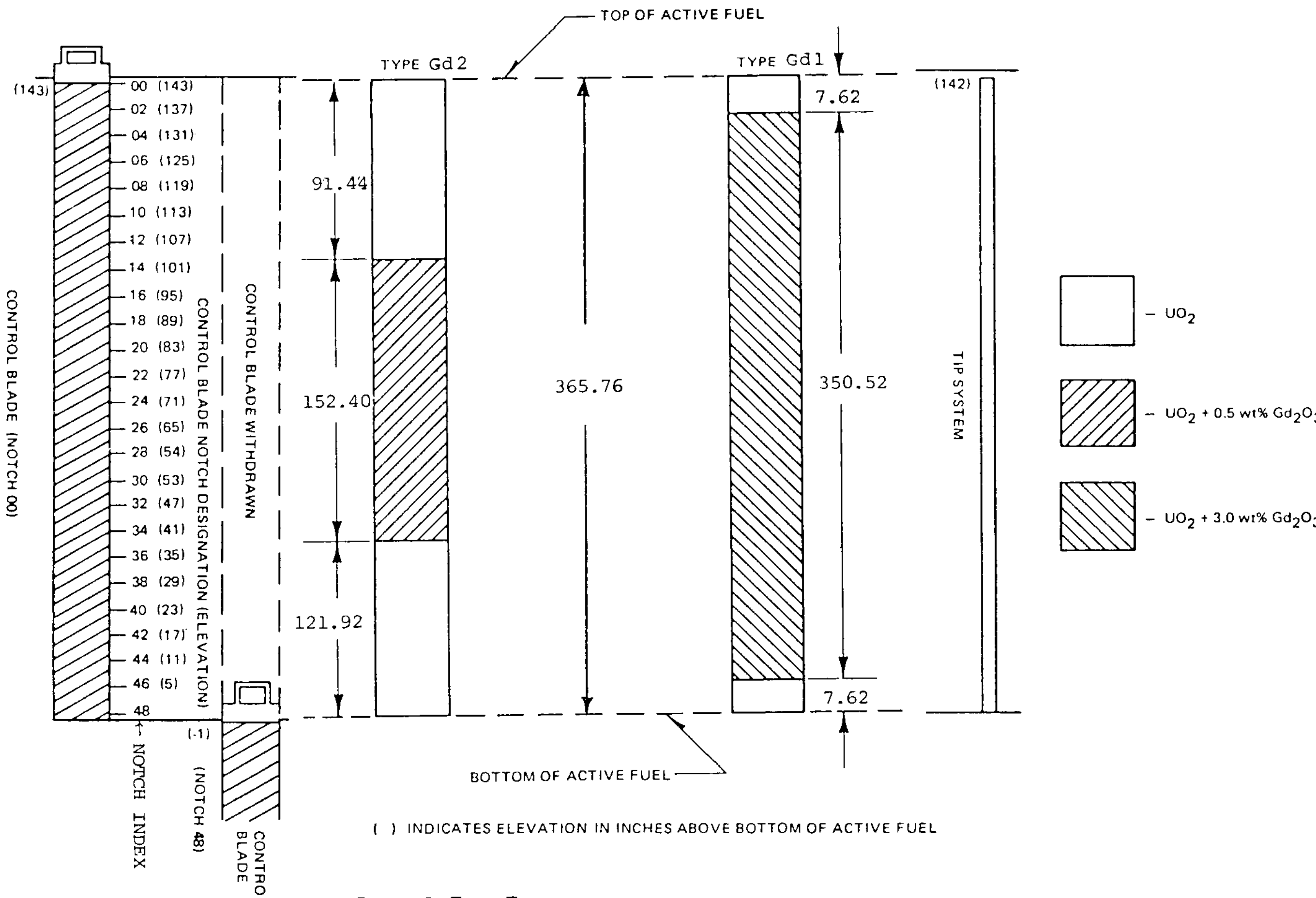


FIG. 6.7 ELEVATION OF CORE COMPONENTS AND AXIAL GD-ZONING OF INITIAL FUEL. (DIMENSIONS IN CM)
QUAD CITIES UNIT TWO 7 X 7 ASSEMBLIES

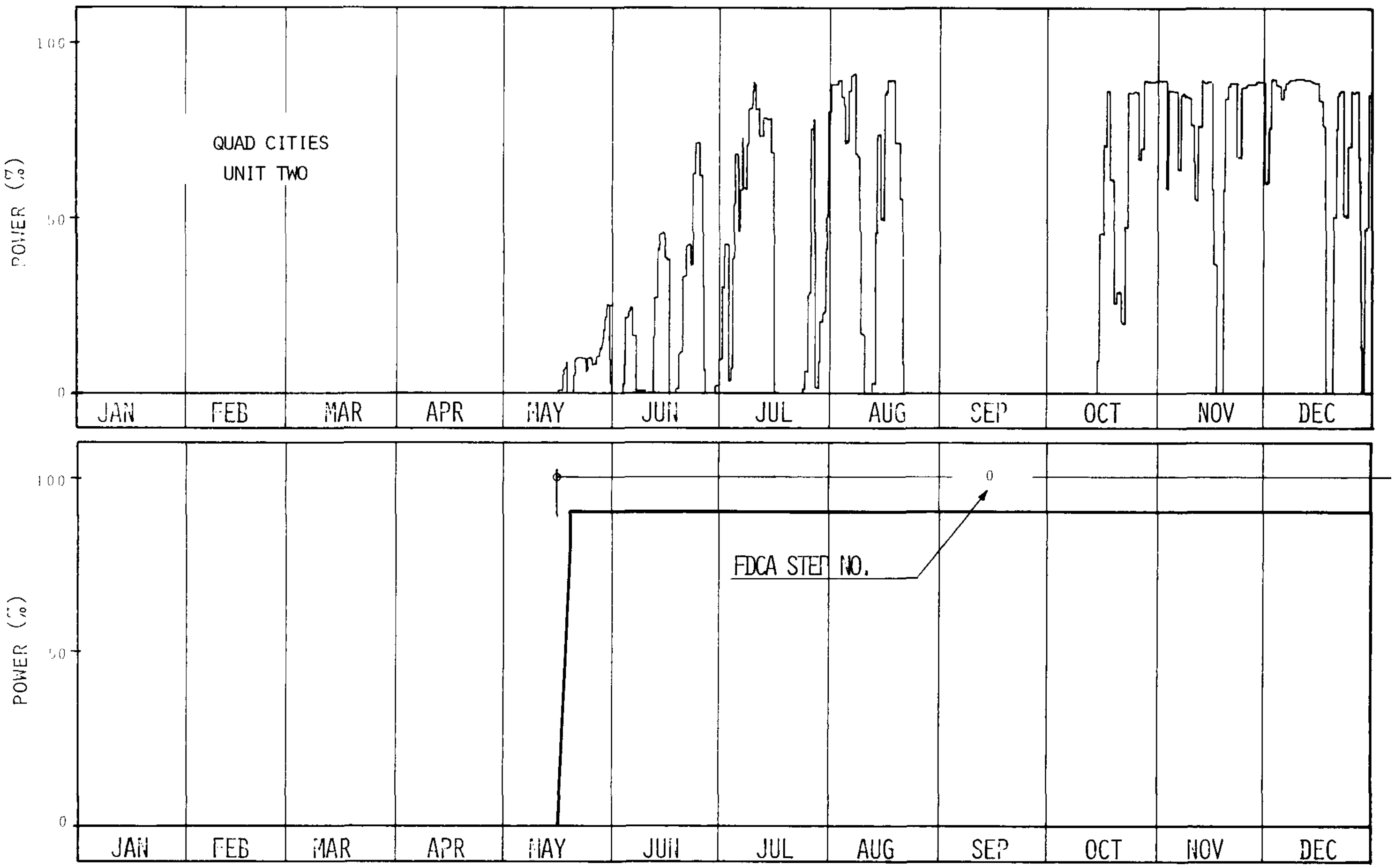
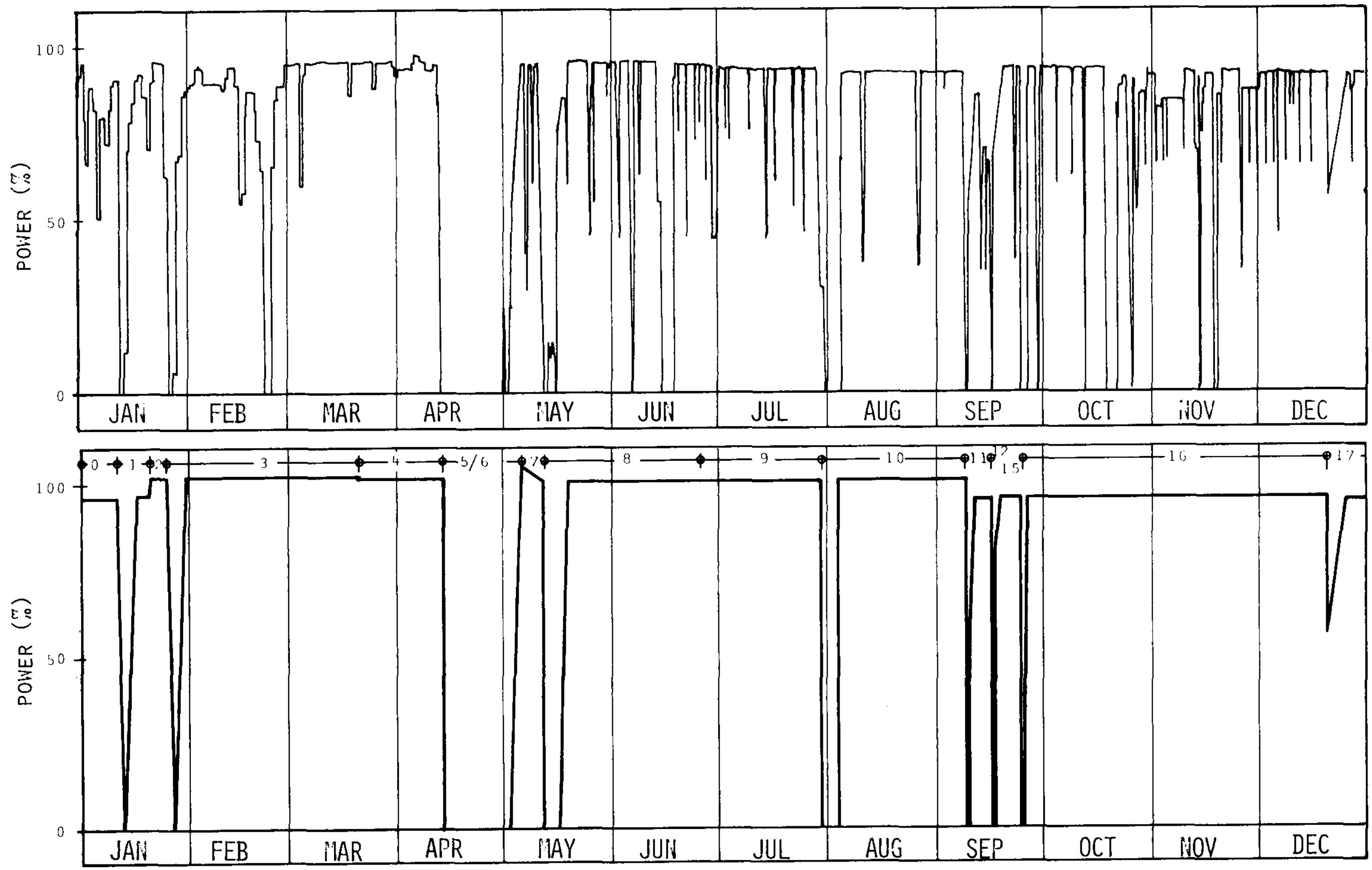


FIG. 6.8 ACTUAL AND SIMULATED REACTOR POWER HISTORY - CYCLE 1. (CONTINUED NEXT PAGE) 1972

QUAD CITIES UNIT TWO



1973

FIG. 6.8 (CONT) ACTUAL AND SIMULATED REACTOR POWER HISTORY - CYCLE 1.

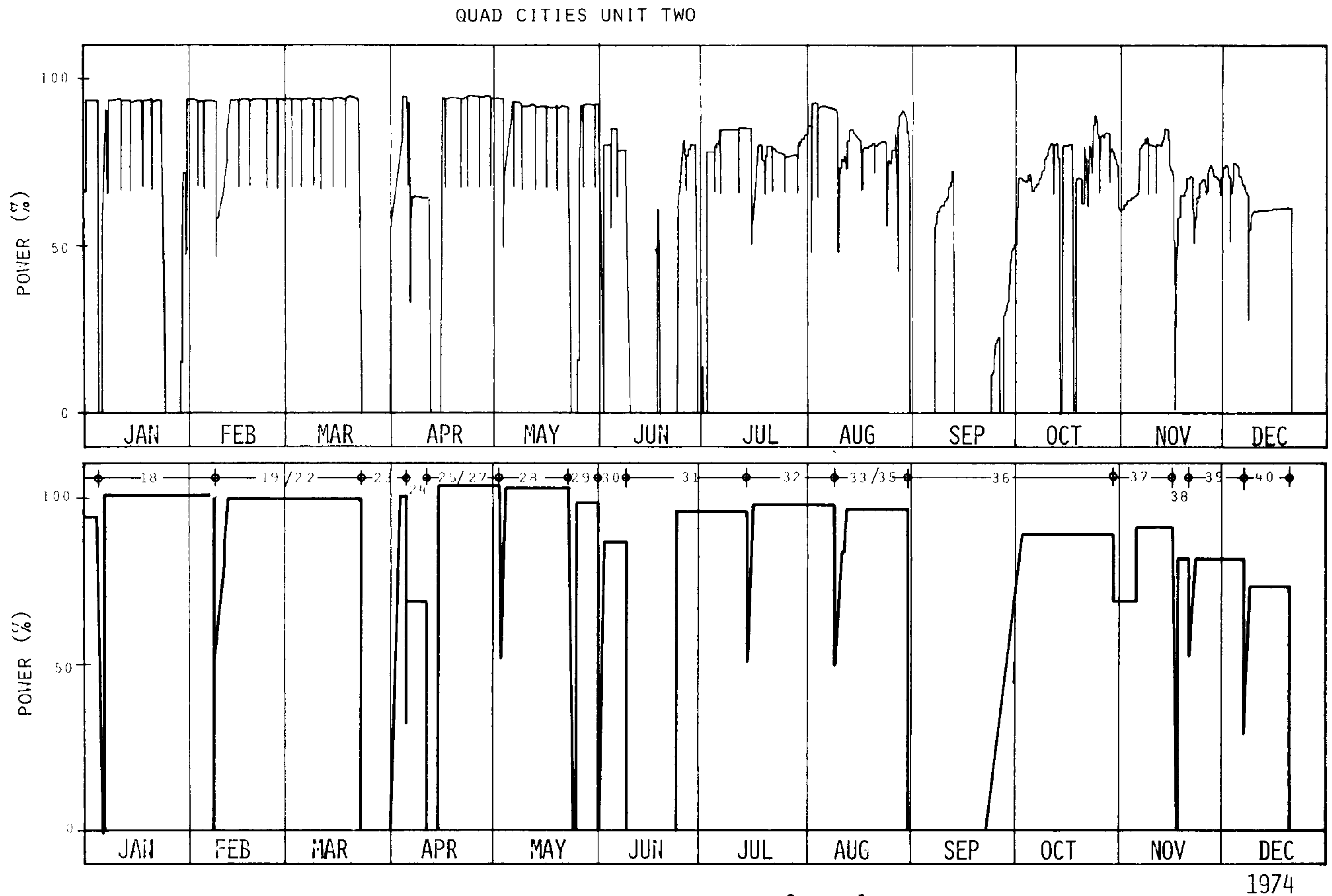


FIG. 6.8 ACTUAL AND SIMULATED REACTOR POWER HISTORY - CYCLE 1

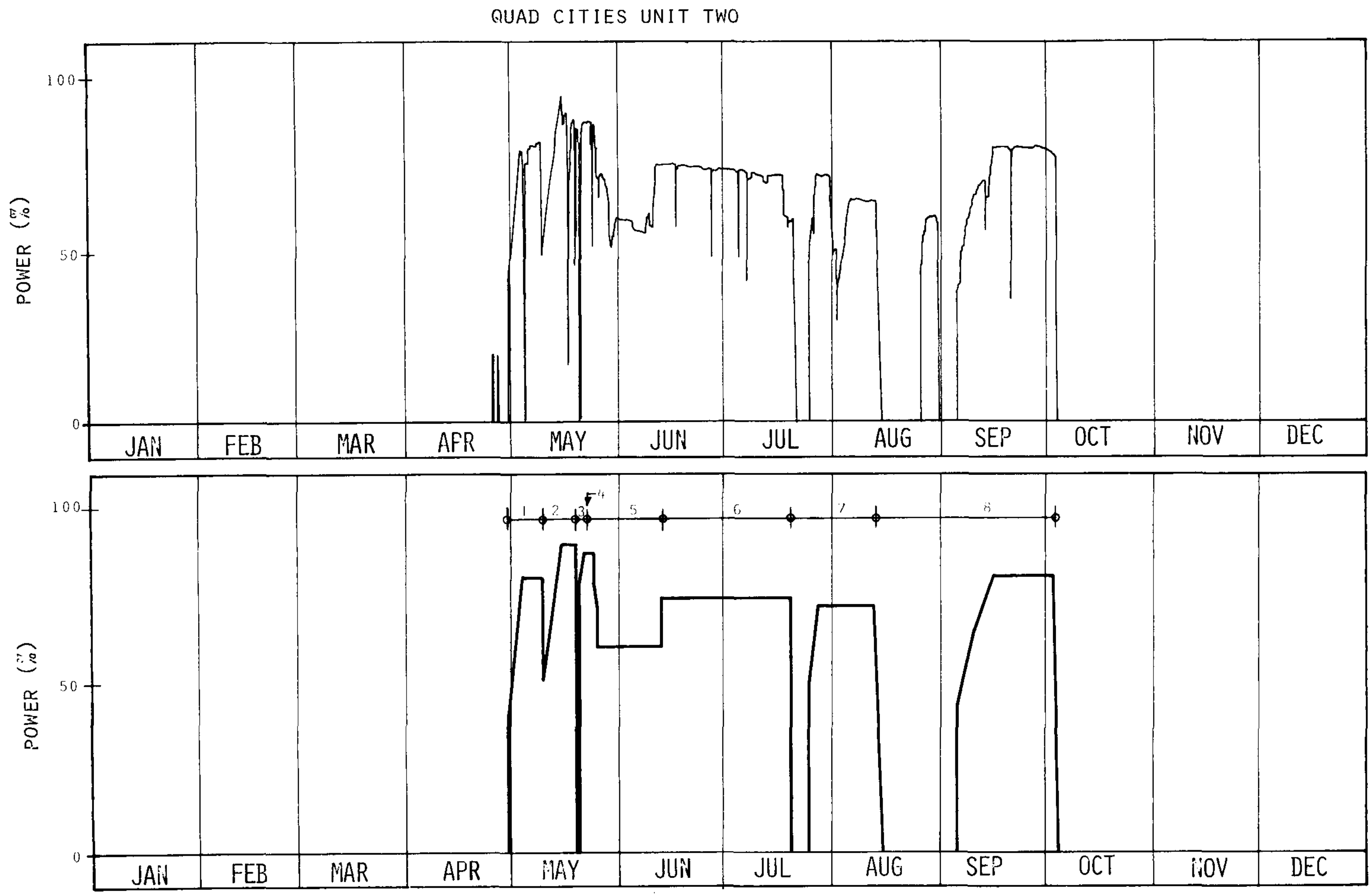


FIG. 6.9 ACTUAL AND SIMULATED REACTOR POWER HISTORY - CYCLE 2

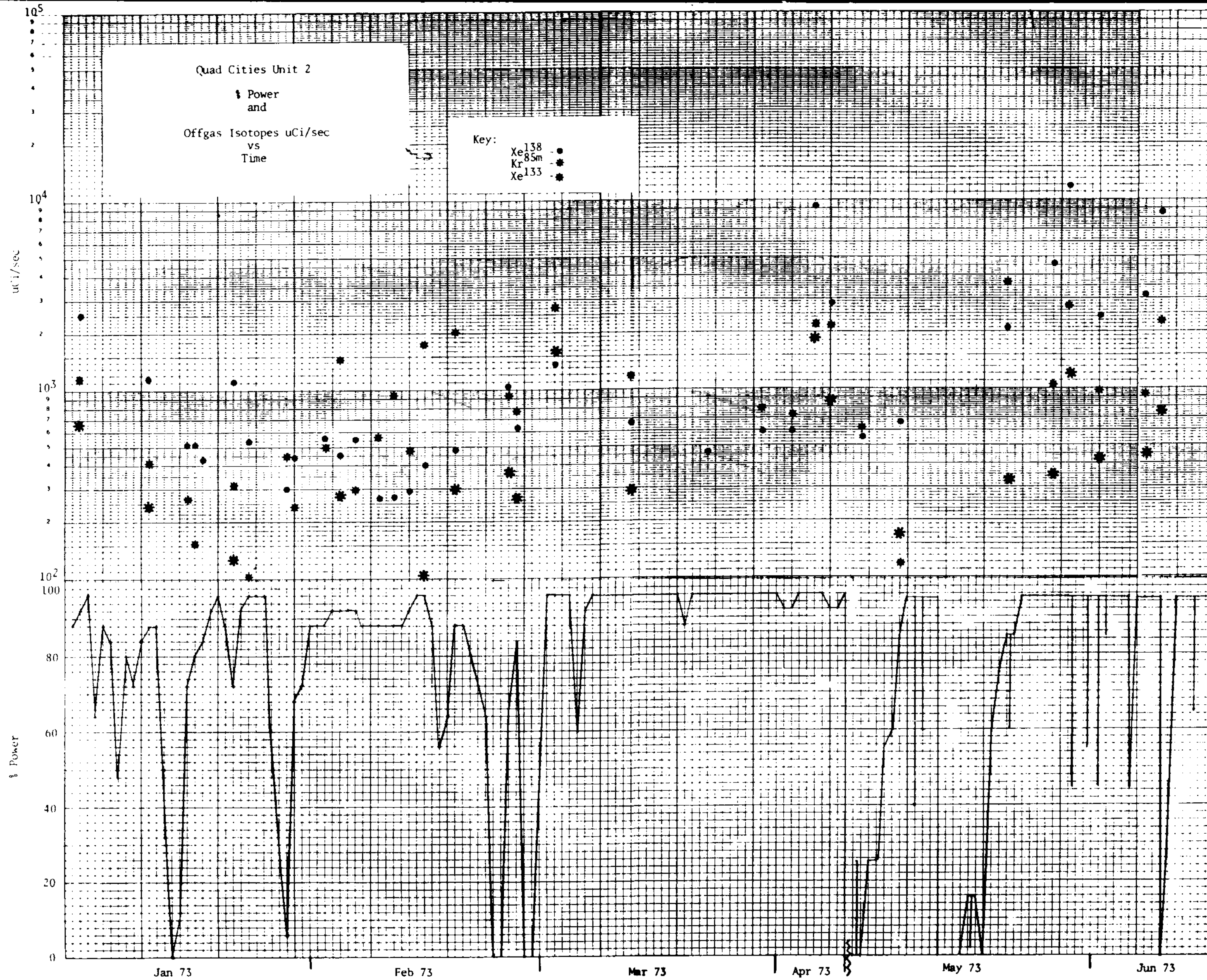


FIG. 6.10 EXAMPLE OF OFF-GAS ACTIVITY PLOT FOR INDIVIDUAL ISOTOPES.

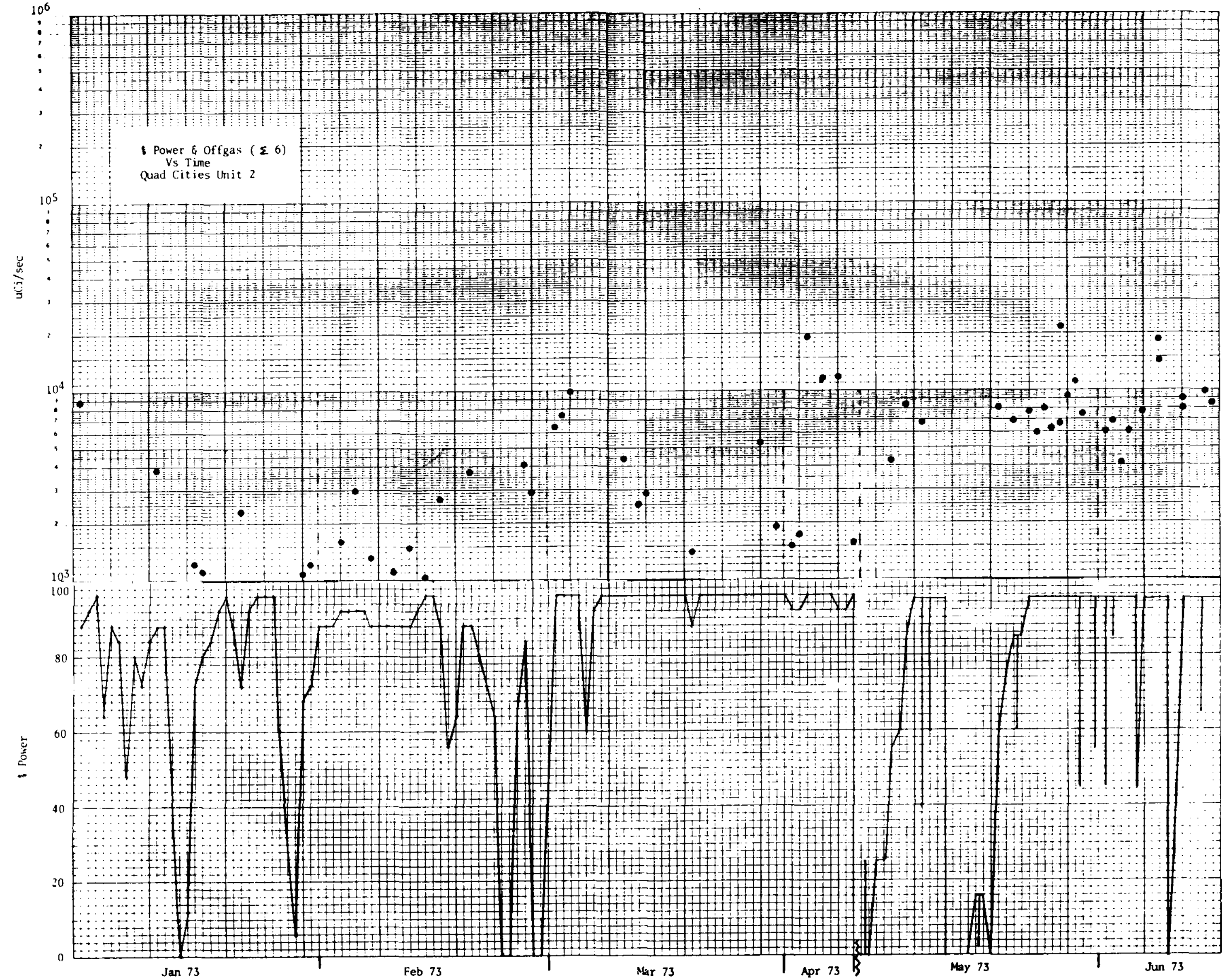


FIG. 6.11 EXAMPLE OF OFF-GAS ACTIVITY PLOT. SUM OF SIX ISOTOPES.

QUAD CITIES UNIT TWO

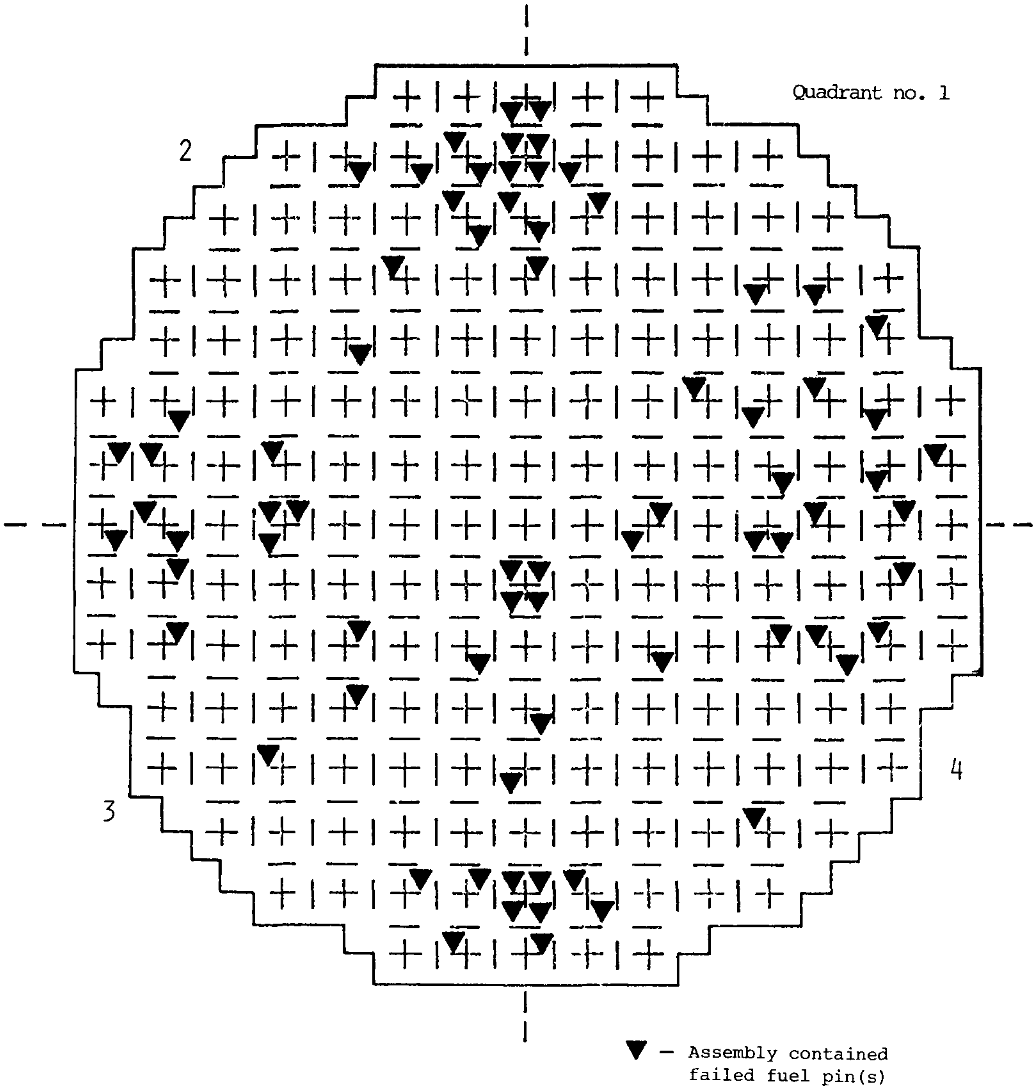


FIG. 6.12 LOCATIONS OF FAILED ASSEMBLIES AS IDENTIFIED BY SIPPING MEASUREMENTS, - CYCLE 1.

QUAD CITIES UNIT TWO

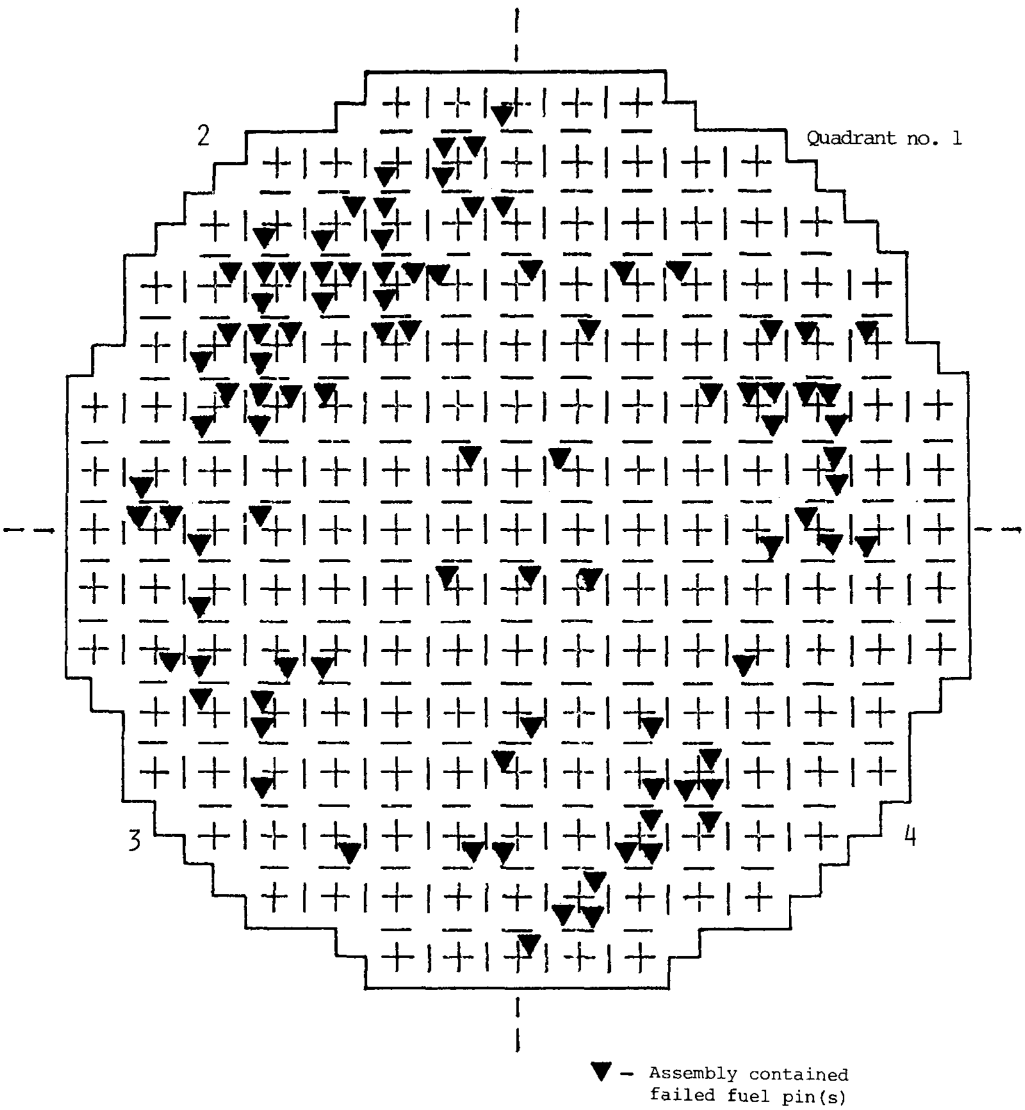


FIG. 6.13 LOCATIONS OF FAILED ASSEMBLIES AS IDENTIFIED BY SIPPING MEASUREMENTS – CYCLE 2.

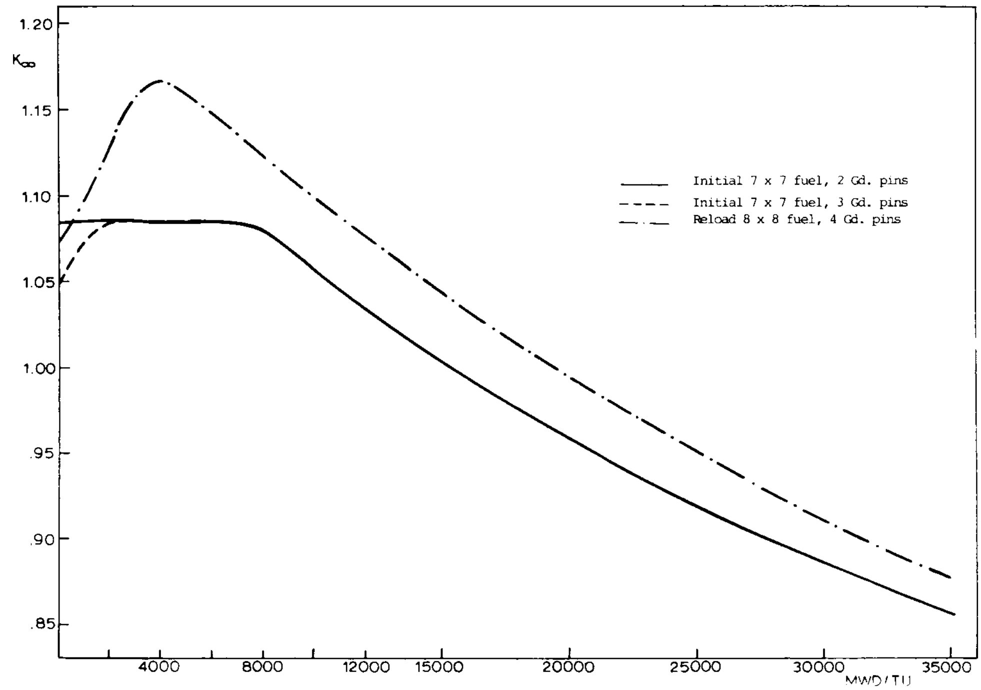


FIG. 6.14 K_∞ VERSUS BURNUP AT 40% VOID FOR INITIAL AND RELOAD FUEL. QUAD CITIES UNIT TWO

Wide-wide Corner

4	4	4	4	4	3	3
4	4	3	3	3	3	3
4	3	2	2	2	2	2
4	3	2	2	1	1	1
4	3	2	1	1	1	1
3	3	2	1	1	1	1
3	3	2	1	1	1	1

← pin-group no.

Group No.	No. Pins	PGR/PGU *	
1	15	1.398	
2	10	1.088	
3	14	0.903	
4	10	0.578	

* PGR = pin-group power dens., rod in
PGU = pin-group power dens., rod out

FIG. 6.15 PIN-GROUP DEFINITION AND RATIO OF RODDED-TO-UNRODDED PIN-GROUP POWER DENSITIES. INITIAL 7 x 7 FUEL.

Wide-wide Corner

4	4	4	4	4	3	3	2
4	4	3	3	3	3	2	2
4	3	3	3	3	3	2	1
4	3	3	3	2	2	2	1
4	3	3	2	H_2O	2	2	1
3	3	3	2	2	2	2	1
3	2	2	2	2	2	2	1
2	2	1	1	1	1	1	1

← pin-group no.

Group No.	No. Pins	PGR/PGU *	
1	11	1.358	
2	22	1.202	
3	20	0.861	
4	10	0.529	

* PGR = pin-group power density, rod in
PGU = pin-group power density, rod out

FIG. 6.16 PIN-GROUP DEFINITION AND RATIO OF RODDED-TO-UNRODDED PIN-GROUP POWER DENSITIES. RELOAD 8 x 8 FUEL.

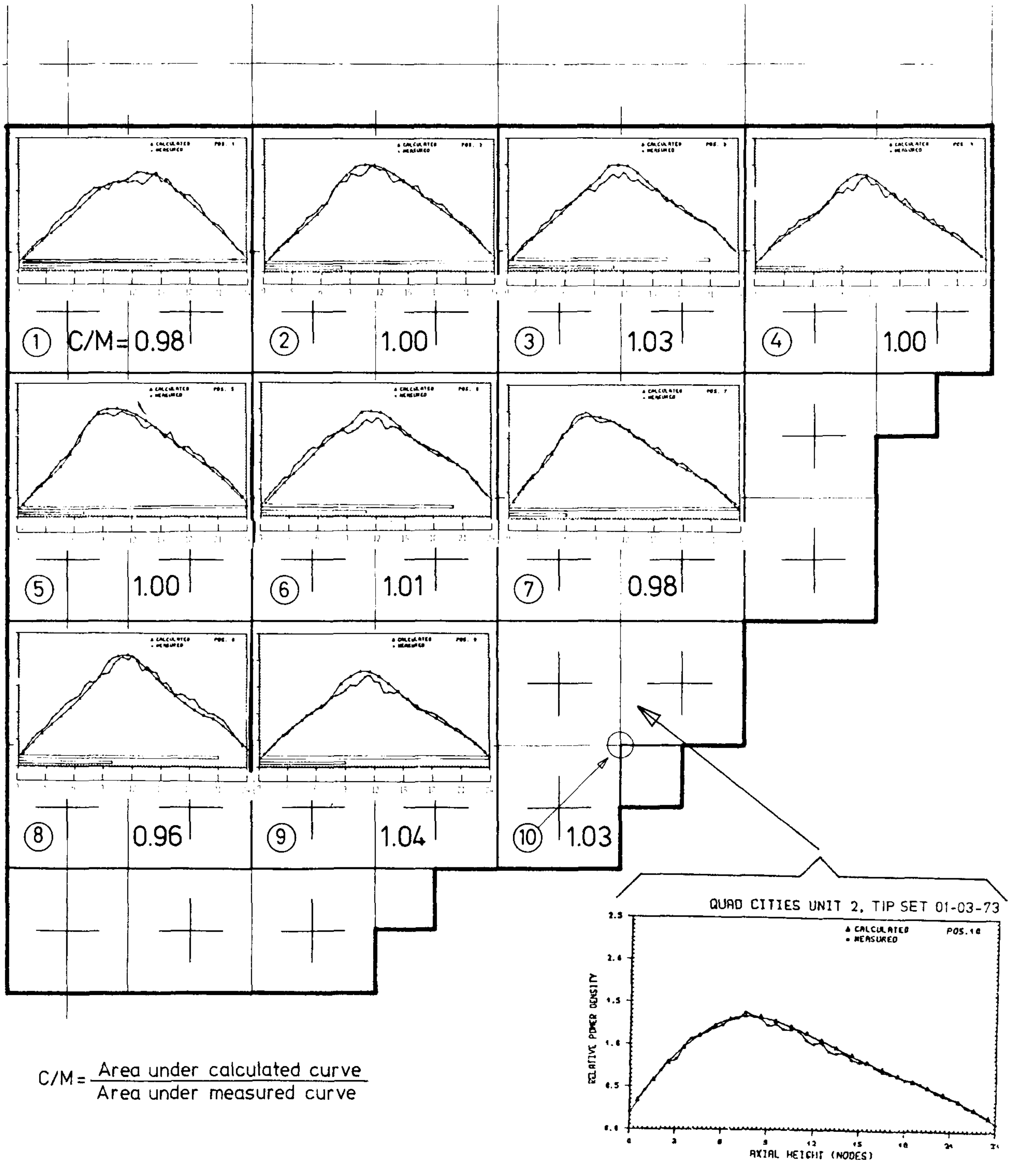


FIG. 6.17 EVALUATION OF CALCULATED POWER DISTRIBUTIONS. COMPARISONS WITH TIP-TRACES - BOC-1.

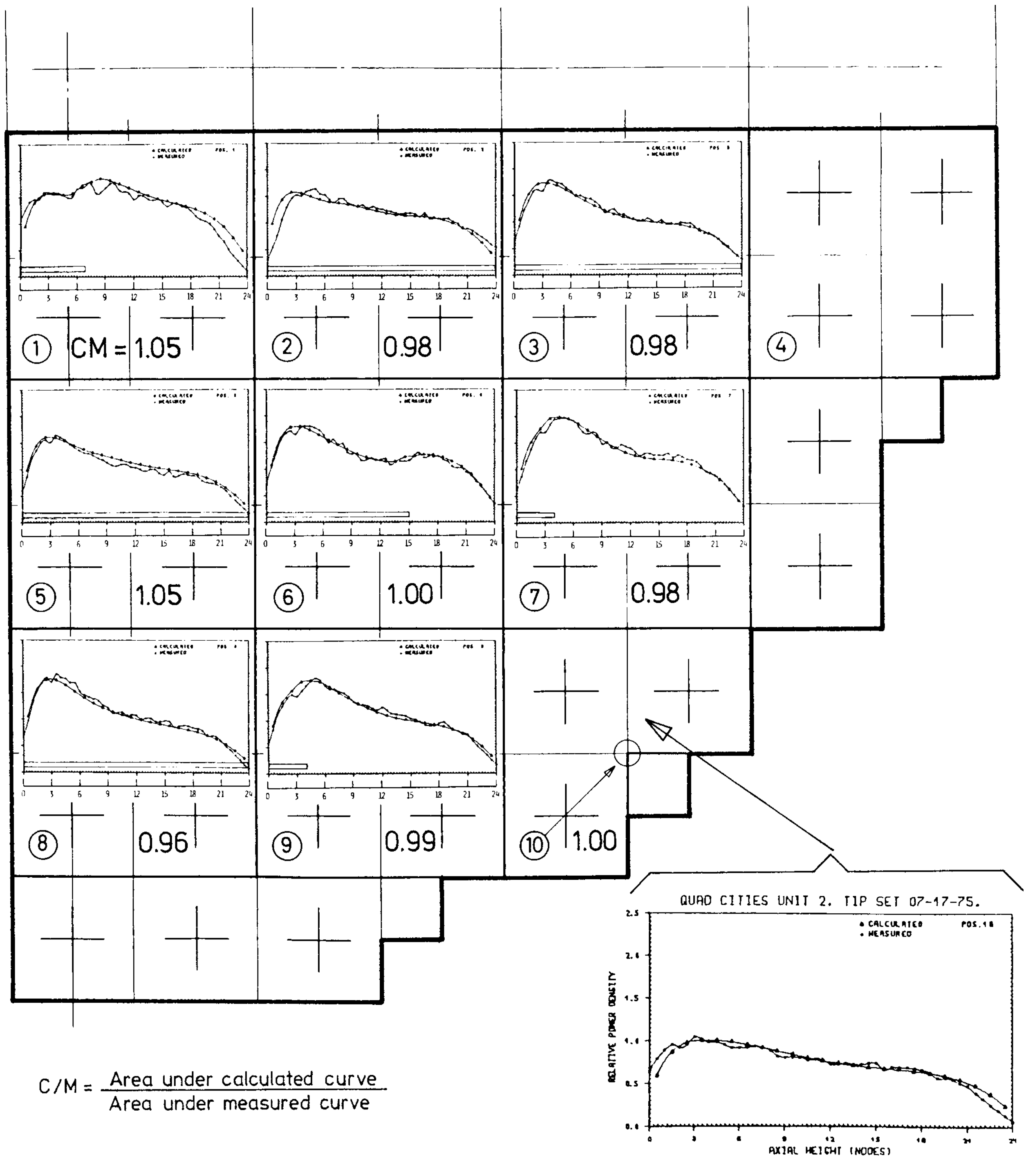


FIG. 6.18 EVALUATION OF CALCULATED POWER DISTRIBUTIONS. COMPARISONS WITH TIP-TRACES - BOC-2.

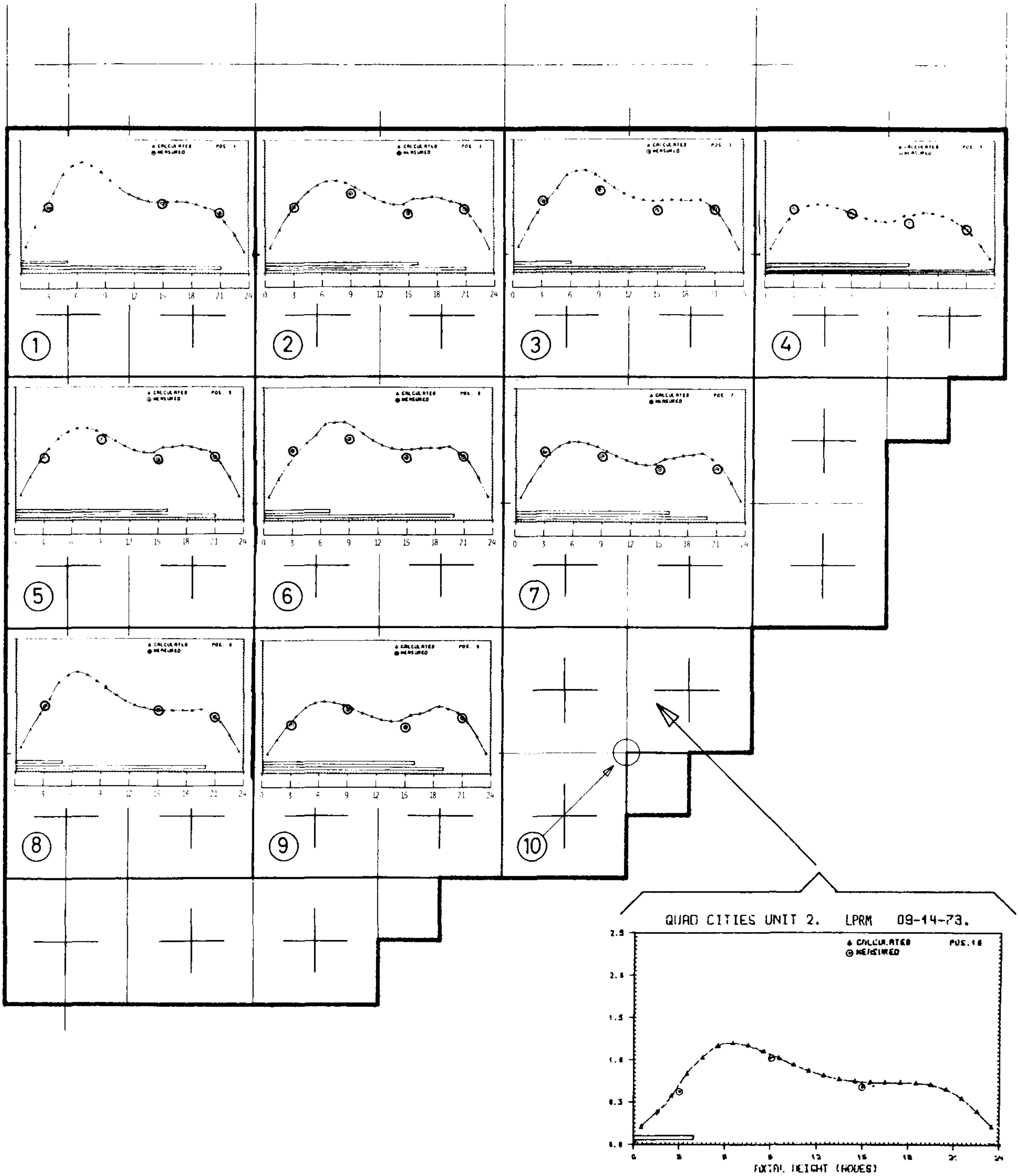


FIG. 6.19 EVALUATION OF CALCULATED POWER DISTRIBUTIONS. COMPARISONS WITH LPRM RECORDINGS.

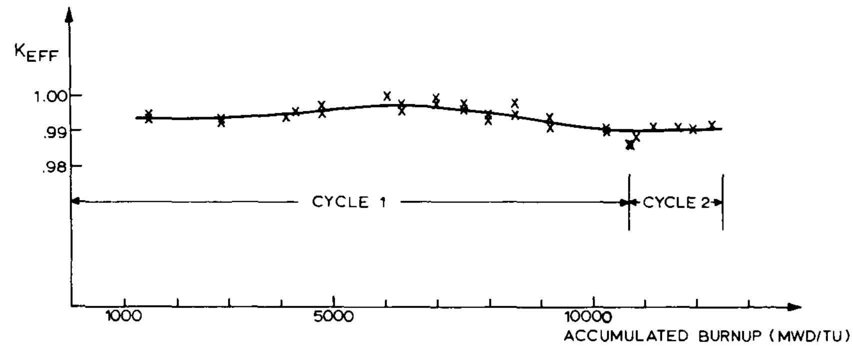


FIG. 6.20 K_{EFF} VERSUS ACCUMULATED CORE AVERAGE BURNUP.

QUAD CITIES UNIT TWO

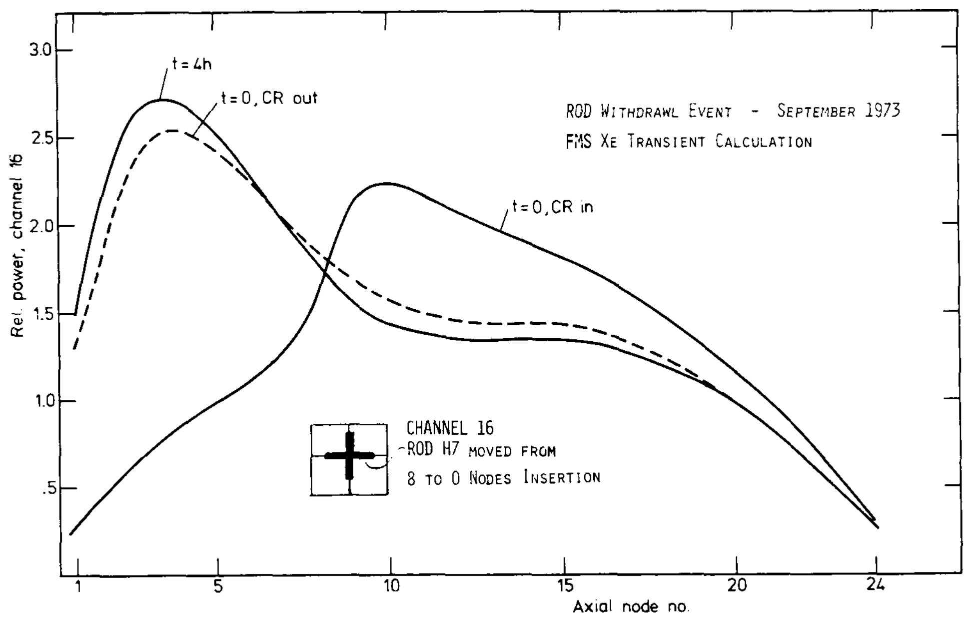


FIG. 6.21 LOCAL AXIAL POWER DISTRIBUTION DURING ROD WITHDRAWAL EVENT - SEPTEMBER 1973.

QUAD CITIES UNIT TWO

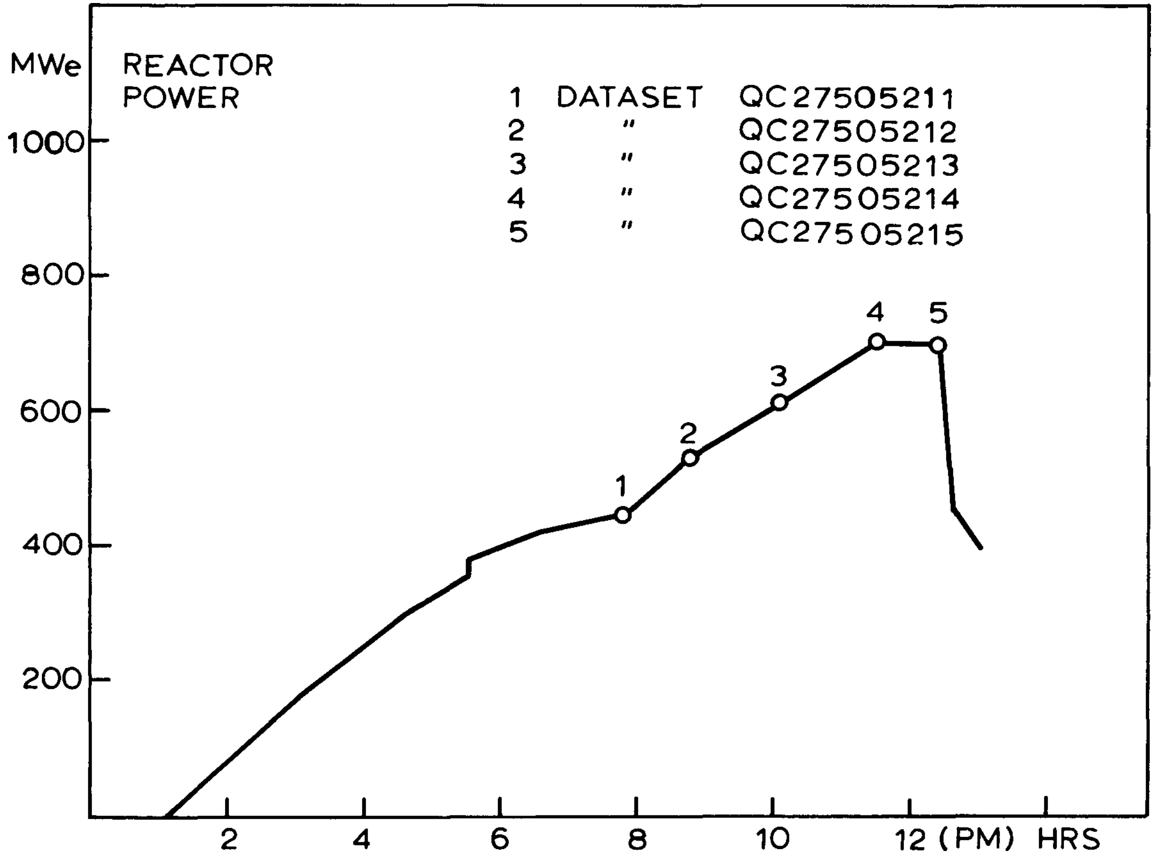


FIG. 6.22 REACTOR POWER VERSUS TIME FOR START UP EVENT ON MAY 22, 1975. REACTOR CONDITIONS REPRESENTED IN THE SIMULATION ARE INDICATED.

QUAD CITIES UNIT TWO

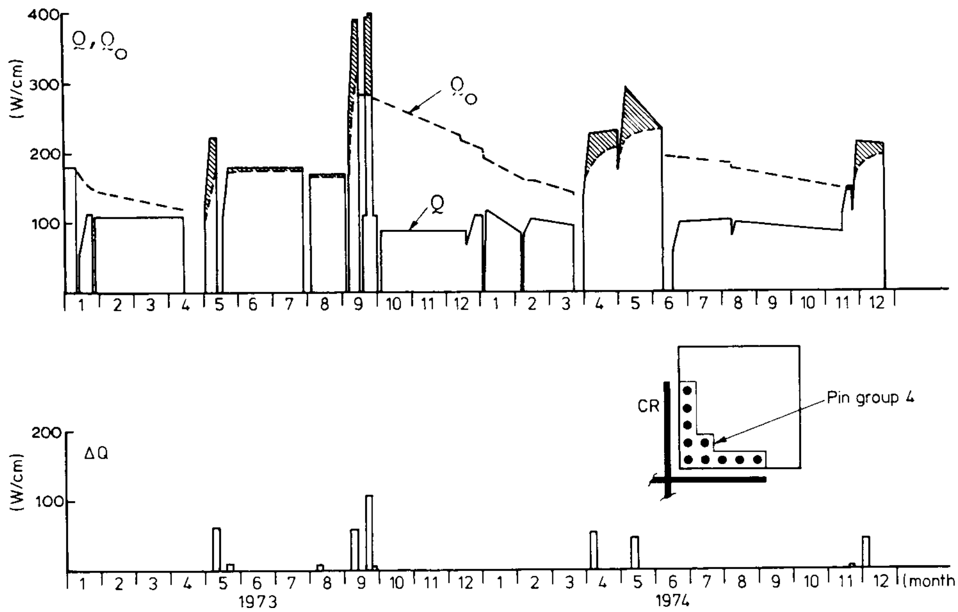


FIG. 6.23 POWER HISTORY (Q), INTERACTION LEVEL (Q_0) AND POWER SHOCK (ΔQ) AS A FUNCTION OF TIME FOR PIN GROUP 4 IN NODE 6 OF ASSEMBLY 903 - CYCLE 1 (1972 NOT SHOWN).

QUAD CITIES UNIT TWO

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FOCA2
****
CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	59	24	19	12	13	29	31	20	19	21	22	13	11	6	0
2	99 ●	20	13	12	14	9	14	14	18	10	10	17	12	5	0
3	99 ⊗	23	13	15	14	10	11	18	18	12	11	15	13	4	0
4	25	20	23	13	14	15	22	10	14	22	29	15	10	4	0
5	14	23	23	17	16	20	21	11	13	31	35	20	8	3	0
6	12	9	12	14	17	8	9	9	12	16	24	25	6	1	
7	14	11	10	14	14	9	9	18	18	15	20	13	2		
8	14	14	18	8	10	17	23	12	15	21	17	7	1		
9	14	18	18	9	9	23	24	17	19	26	14	3	0		
10	13	7	9	10	12	11	14	29	21	11	5	1			
11	13	8	8	13	13	9	11	21	12	5	1				
12	19	35	30	11	8	7	6	6	4	1					
13	18	37	32	10	5	4	2	1	0						
14	6	6	5	3	2	0									
15	1	1	0	0	0										

RWE - event:
● 903
⊗ 302

Identification of pos. of failed assemblies :

FIG. 6.24 CORE-WIDE PREDICTION OF PROBABILITY FOR ASSEMBLY FAILURES. COMPARISONS WITH FAILURES IDENTIFIED BY SIPPING EOC-1.

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA2
****

CASE TITLE = F2-OC-02-00 QUAD CITIES - 2 , CYCLE 2, 04.

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	10	10	8	9	8	5	8	9	11	9	8	21	8	3	0
2	13	0	15	0	3	0	9	0	8	0	8	0	13	0	0
3	9	15	8	4	3	4	7	9	6	4	4	7	5	3	C
4	9	0	4	0	5	0	6	0	8	0	9	0	5	0	0
5	9	4	4	6	6	4	6	10	4	8	5	4	2	0	0
6	7	0	4	0	6	0	2	0	9	0	24	0	3	0	
7	8	7	4	9	15	4	7	10	8	93	15	3	0		
8	10	0	12	C	16	0	12	0	9	0	7	0	0		
9	10	12	12	18	34	40	23	15	11	6	3	0	0		
10	14	0	5	0	46	0	65	0	12	4	0	0			
11	10	7	7	24	37	42	28	12	7	0	0				
12	19	0	17	0	30	0	9	0	1	0					
13	9	14	20	6	9	4	0	0	0						
14	10	0	6	0	1	0									
15	0	0	0	0	0										

○ - Assembly identified as failed by sipping.

FIG. 6.25 ASSEMBLY FAILURE PROBABILITIES FOR QUADRANT 4 - COMPARISON WITH FAILURES IDENTIFIED BY SIPPING EOC-2.

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA2
****
CASE TITLE = F2-QC-02-00 QUAD CITIES - 2 , CYCLE 2, C2.

FAILURE PROBABILITY IN ASS. * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	9	9	8	7	8	4	8	10	11	11	8	28	8	4	0
2	13	0	15	0	3	0	9	0	10	0	9	0	15	0	0
3	9	15	8	3	2	4	7	12	11	5	6	12	11	4	0
4	8	0	4	0	5	0	8	0	16	0	26	0	6	0	0
5	8	3	3	5	6	5	9	16	26	35	21	20	7	0	0
6	5	0	4	0	5	0	4	0	31	0	38	0	4	0	
7	7	7	4	8	14	5	9	21	46	85	24	8	0		
8	6	0	9	0	15	0	18	0	49	0	24	0	0		
9	9	8	8	14	30	43	34	34	20	30	12	0	0		
10	9	0	4	0	38	0	71	0	29	6	2	0			
11	9	6	5	17	28	37	29	15	11	1	0				
12	11	0	12	0	21	0	8	0	1	0					
13	8	11	12	5	7	4	0	0	0						
14	5	0	4	0	0	0									
15	0	0	0	0	0										

○ - Assembly identified as failed by sipping.

FIG. 6.26 ASSEMBLY FAILURE PROBABILITIES FOR QUADRANT 2 - COMPARISON WITH FAILURES IDENTIFIED BY SIPPING EOC-2.

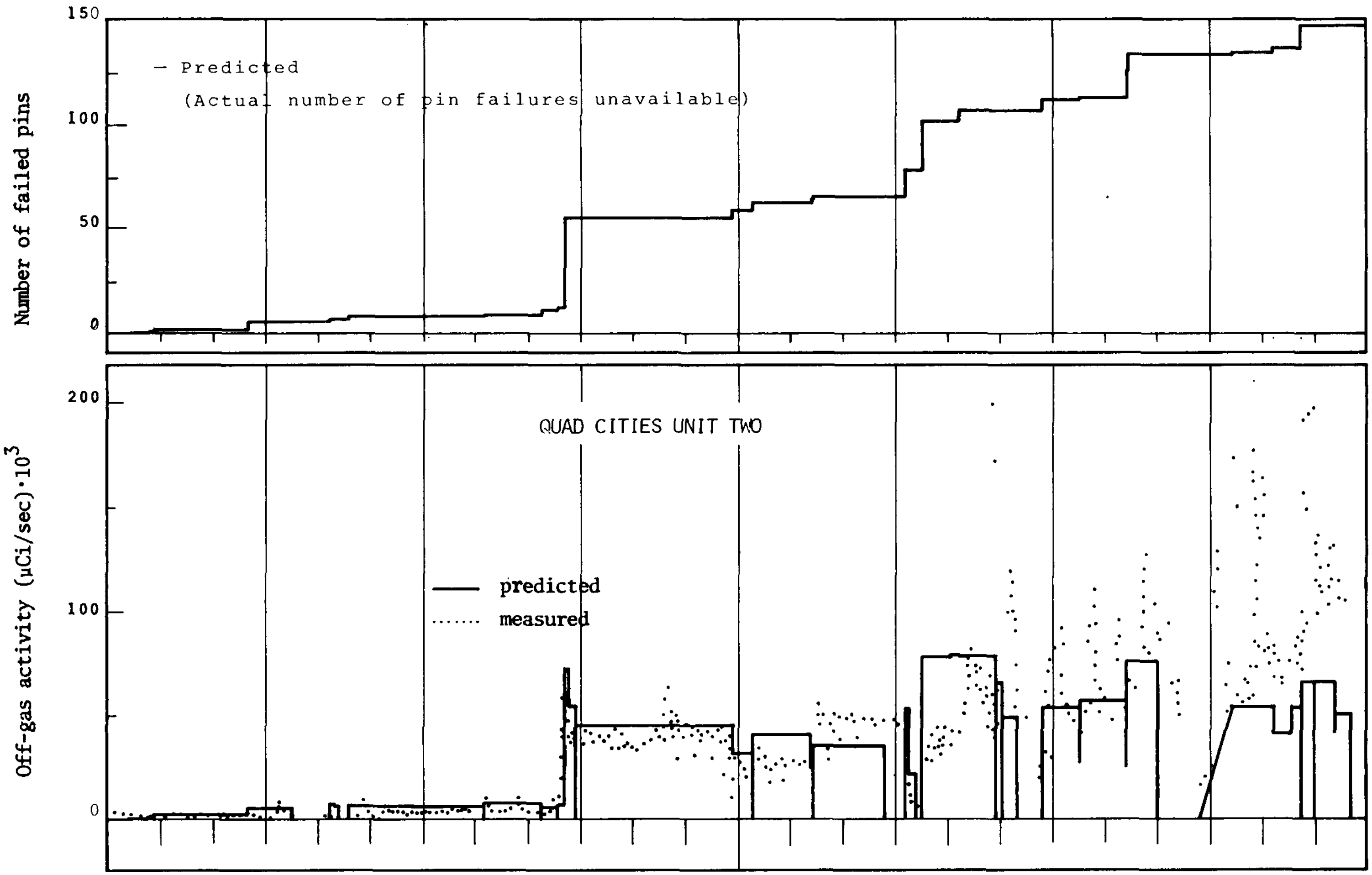


FIG. 6.27 PREDICTED ACCUMULATED PIN FAILURES AND MEASURED/PREDICTED OFF-GAS ACTIVITY LEVEL VERSUS TIME - CYCLE 1.

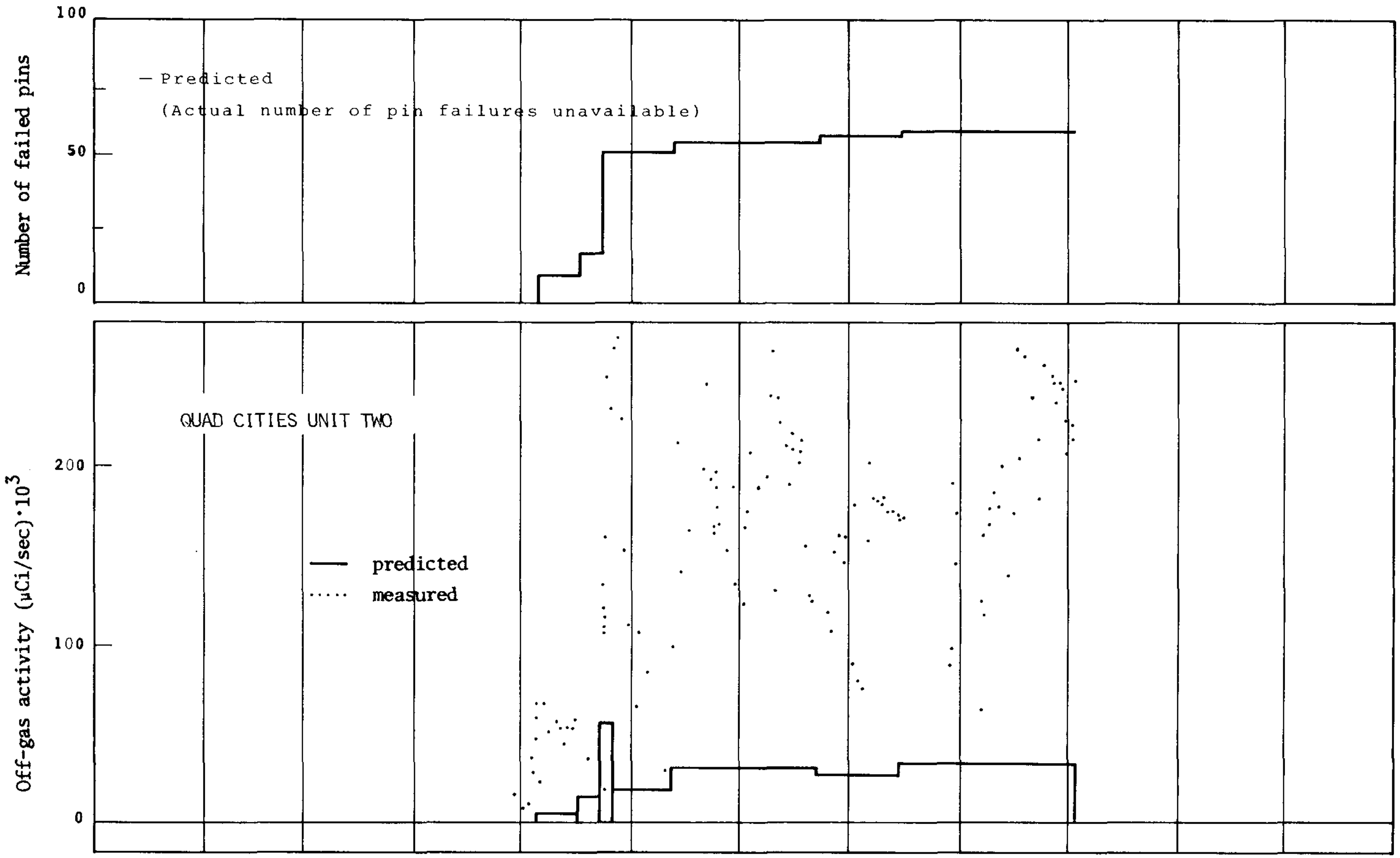


FIG. 6.28 PREDICTED ACCUMULATED PIN FAILURES AND MEASURED/PREDICTED OFF-GAS ACTIVITY LEVEL VERSUS TIME - CYCLE 2.

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA2

****

CASE TITLE = F2-OC-01-00 QUAD CITIES - 2 , CYCLE 1

NUMBER OF FAILED PINS IN ASS. 1[illegible].

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	89	28	21	13	13	34	37	23	21	24	25	14	12	6	0
2	1084 ●	22	14	13	16	10	15	16	19	10	11	19	12	5	0
3	892 ⊗	26	14	16	15	10	12	19	19	13	11	16	14	4	0
4	28	22	26	14	16	16	24	11	15	25	35	16	11	4	0
5	15	26	26	19	18	22	24	12	14	38	43	22	9	3	0
6	13	10	13	15	19	8	10	10	13	18	27	29	6	1	
7	15	12	11	15	15	9	10	19	20	16	22	14	2		
8	15	15	20	8	11	19	26	12	17	24	19	7	1		
9	16	20	20	9	10	26	28	19	21	30	15	3	0		
10	14	7	9	10	13	11	15	35	24	12	5	1			
11	14	9	9	14	14	9	12	24	13	5	1				
12	21	44	36	12	9	7	6	6	4	1					
13	21	47	38	11	5	4	2	1	0						
14	6	6	5	3	2	0									
15	1	1	0	0	0										

RWE - EVENT:

● 903

⊗ 302

FIG. 6.29 CORE-WIDE PREDICTION OF MOST PROBABLE NUMBER OF FAILED PINS - EOC-1.

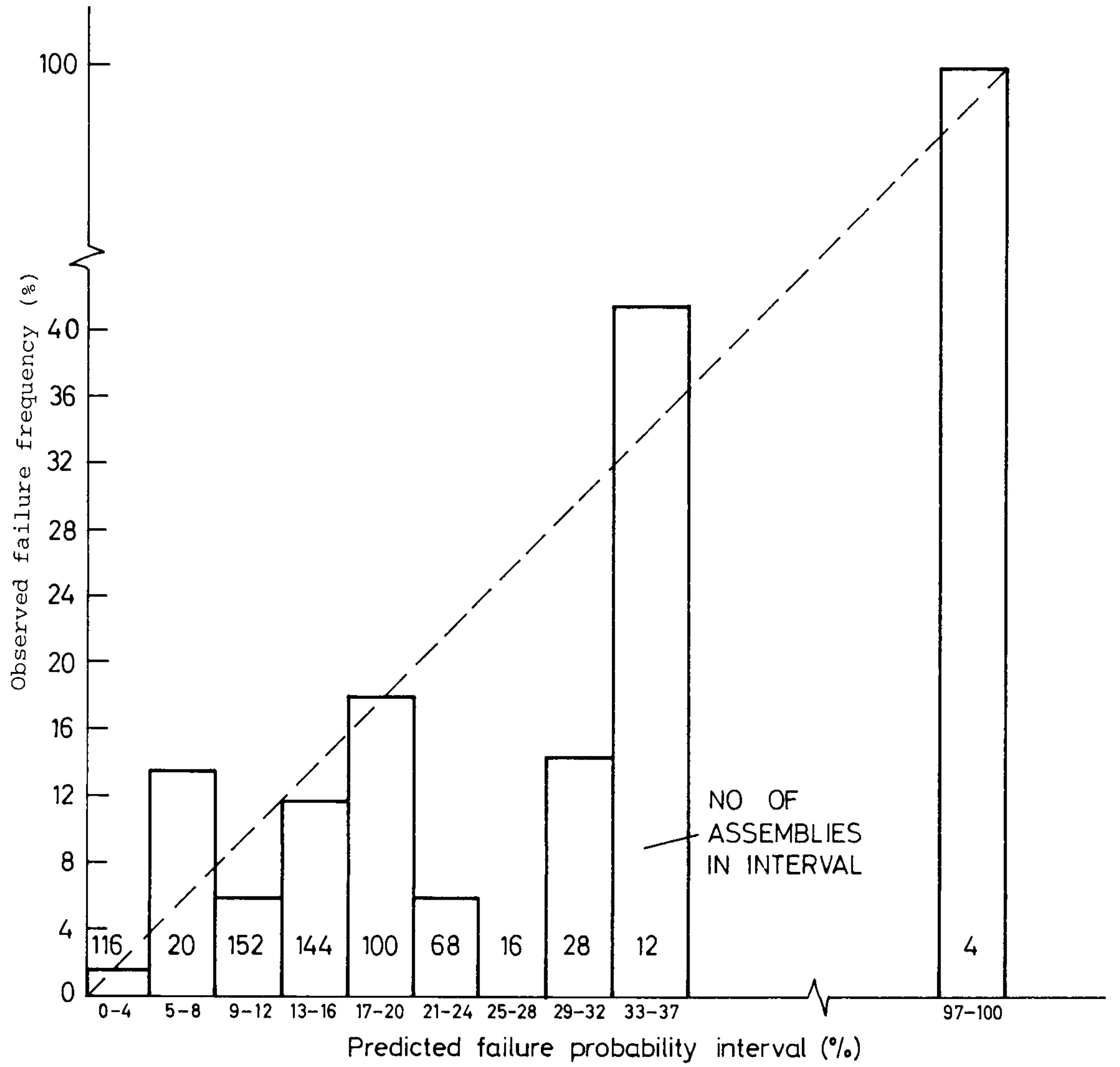
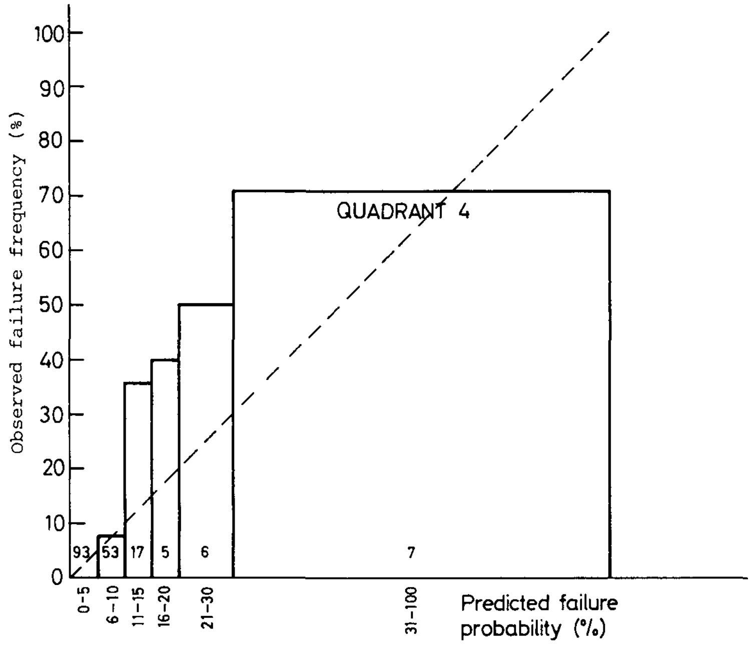


FIG. 6.30 ERROR PLOT OF PREDICTED FAILURE PROBABILITY VERSUS OBSERVED. CYCLE 1 - QUAD CITIES 2. DATA FROM FIG. 6.24



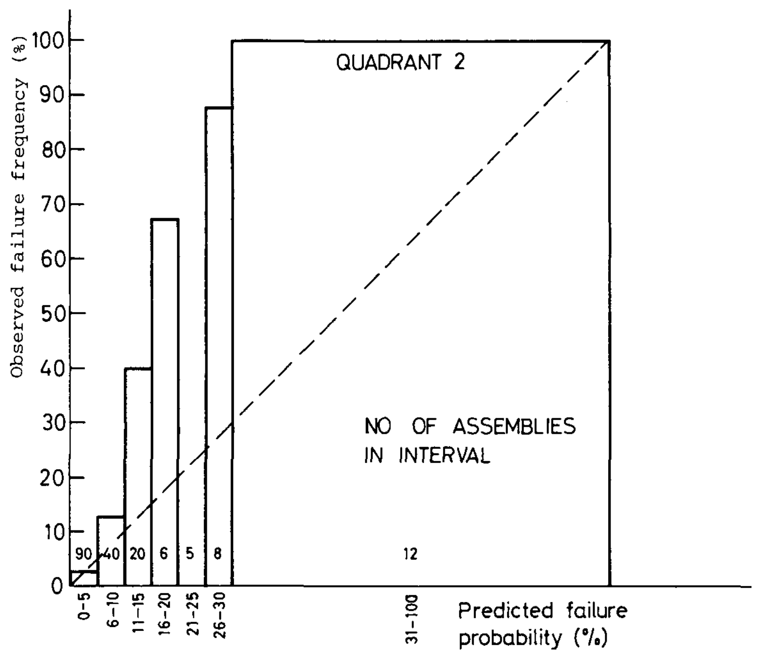


FIG. 6.31 ERROR PLOTS OF PREDICTED FAILURE PROBABILITY VERSUS OBSERVED. QUAD CITIES 2 - CYCLE 2.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	24/10	25/10	1/8	13/9	5/8	29/5	1/8	12/9	0/11	22/9	1/8	13/21	0/8	10/3	13/0
2	14/13		21/15		14/3		14/9		13/8		2/8		0/13		10/0
3	0/9	9/15	13/8	15/4	14/3	10/4	11/7	13/9	19/6	20/4	15/4	5/7	24/5	4/3	29/0
4	25/9		13/4		17/5		14/6		6/8		6/9		0/5		13/0
5	5/9	23/4	23/4	17/6	23/6	20/4	7/6	3/10	23/4	31/8	9/5	20/4	8/2	11/0	20/0
6	9/7		8/4		17/6		12/2		11/9		17/24		1/3	25/0	
7	1/8	10/7	10/4	14/9	6/15	15/4	6/7	3/10	18/8	0/93	18/15	16/3	15/0		
8	14/10		18/12		4/16		4/12		21/9		6/7		14/0		
9	0/10	18/12	12/12	6/18	12/34	9/40	9/23	20/15	2/11	26/6	14/3	19/0	24/0		
10	13/14		14/5		12/46		1/65		21/12	0/4	12/0	23/0			
11	1/10	2/7	30/7	10/24	13/37	11/42	23/28	7/12	12/7	14/0	15/0				
12	7/19		5/17		8/30		11/9		23/1	11/0					
13	0/9	1/14	12/20	0/6	5/9	1/4	10/0	13/0	14/0						
14	18/10		3/6		9/1	8/0									
15	32/0	37/0	13/0	21/0	11/0										

Quadrant 4

Assembly Failed During Cycle 2

Open boxes represent reload assemblies

FIG. 6.32 ASSEMBLY FAILURE PROBABILITIES EOC-1/EOC-2 - %

QUAD CITIES UNIT TWO

Section 7

MAINE YANKEE ANALYSIS

DESIGN AND PERFORMANCE DATA

Maine Yankee is an 830 MWe Combustion Engineering, Inc., (CE) PWR, operated by the Maine Yankee Atomic Power Company. Operation started in November 1972. The initial core loading consisted of 69 Type A assemblies and 80 Type B assemblies, arranged in a checkerboard pattern, surrounded by 68 Type C assemblies in the peripheral core region. Type B, and some of the Type C assemblies, contained burnable poison in the form of boron carbide shim rods. The initial loading consisted of unpressurized fuel pins loaded with fuel pellets which were susceptible to in-reactor densification.

Of the 217 assemblies, 37 carried fixed power distribution detectors in central guide tubes. There were 4 detectors axially in each tube. In addition, there were 8 moveable fission chambers. All detectors in the interior (checkerboard) core region were located in Type B assemblies.

The reactor was operated at approximately 80% of nominal full power throughout Cycle 1.

The reactor was shut down for fuel inspection on 29 June 1974. As a result, 72 assemblies were replaced to form Core 1A. The plant returned to commercial operation on 11 October 1974. The 72 fresh fuel assemblies loaded are denoted as Types RF0, RF4 and RFB. The cycle was terminated on 2 May 1975, followed by fuel inspection of the entire Core 1A loading.

The fuel inspection (wet sipping) of the Core 1 loading identified one Type A, 41 Type B and one Type C assemblies as leakers. Out of the Core 1A loading, three Type A, seven Type B and two Type C assemblies were found to be leaking.

Descriptions of the design and performance data used in this study are given below.

Fuel and Assembly Descriptions

- Geometric design data for the fuel pellets, fuel rods and fuel assemblies for the fuel types used in Cycles 1 and 1A are given in Table 7-1.
- Corresponding material composition data and the number of fuel rods and assemblies of each type are given in Table 7-2.
- Shim rod data are given in Table 7-3.
- The fuel assembly layout for each type of shim rod arrangement is shown in Figure 7-1.

Control Rod Design

- The design of the Control Element Assemblies (CEA) is shown schematically in Figure 7-2.
- Absorber rod data are given in Table 7-4.

Core Design

- Table 7-5 gives the total number of assemblies in the core, dimensions, heat transfer area, flow area, etc.
- The core arrangement for Cycle 1, showing the fuel type, layout and the control rod group definition, is shown in Figure 7-3.
- The core arrangement for Cycle 1A is shown in Figure 7-4.

Nominal Operating Conditions

- Nominal values for core thermal power, moderator temperature and flow rate, are given in Table 7-6.

Operating History

- The gross thermal power production history through Cycles 1 and 1A is shown in Figures 7-5 and 7-6, together with the simplified power history, as modeled for this study.
- Lists of operating condition data, giving thermal power, moderator boron content and rod bank positions for a number of selected dates through Cycles 1 and 1A, are given in Tables 7-7 and 7-8.
- Plots of the I^{131} coolant activity level, on a day-by-day basis, were obtained. Simplified plots are given, together with results of the calculations, in Figures 7-18 and 7-19.
- Figure 7-7 shows the locations of the assemblies identified as leakers by the sipping analysis after Cycle 1.
- Figure 7-8 shows the locations of the assemblies identified as leakers by the sipping analysis after Cycle 1A.

FUEL DUTY CYCLE SIMULATION

Nuclear Data Bank Generation

A nuclear data bank for the Maine Yankee core was generated, using the RECORD code. This data bank contains pin power distributions, and 2-group macroscopic cross-sections and diffusion coefficients condensed from the detailed multigroup calculations for each fuel type. These data are generated as a function of fuel burnup and, in the case of a PWR, as a function of boron concentration in the moderator. The burnup calculations were performed up to 35000 MWD/TU, with an average boron concentration of 400 ppm. Using, subsequently, a restart option in RECORD, the burnup-dependent group data and pin power distributions at zero and 800 ppm boron concentration were generated from the 400 ppm calculations. The RECORD calculations also include treatment of shim rods which may be present, and the calculation of depletion of burnable absorber (boron-10) in each such rod.

The main data bank is generated for the fuel at this hot operating condition, with control elements out. Separate calculations were made with RECORD to determine the coefficients required by PRESTO to take into account the effects of Doppler coefficient, variation of moderator density, steady state and transient xenon, and insertion of control elements. Both boron-based and Ag-In-Cd-based control elements can be treated directly by RECORD.

Figure 7-9 shows curves of k_∞ versus burnup, as calculated by RECORD, for the different fuel types in the Maine Yankee core with 400 ppm boron concentration in the moderator. An example of pin power distribution for a 1/4-assembly of Fuel Type A is shown in Figure 7-10. This figure shows also the subdivision into pin groups for the power-shock model which is used for all the different fuel types. Table 7-9 lists the number of pins per pin group for each assembly type. Table 7-10 gives the pin-group power distributions for each fuel type analyzed.

Core Model

The operation of the reactor was simulated with PRESTO, using 1/8-core symmetry. Each of the 217 assemblies was divided into four nodes in the x-y plane and into 24 nodes axially. This results in 2736 nodes in a core octant, including the nodes on the diagonal symmetry line, in PRESTO. Five different fuel types were defined for the initial core, using a separate treatment for each number of shim rods in Fuel Type C (Figure 7-3). The second core (called core 1A) was modeled with 7 different fuel types where Fuel Type 6 was fresh, without shim rods (FO and FB), and Type 7 was fresh, with four shim rods (F4 and AS) (Figure 7-4). The two other fuel types used in the second core were not taken into account, as they appeared in only one and two of the octants, respectively. Fuel shuffling from inside and outside the octant was accounted for. Shuffling from outside the octant was simulated by defining a new assembly, with history from the assembly symmetric to the one shuffled. Rotation of an assembly was simulated by reordering the four parts into which the assembly was divided.

The reactor operating condition, for a given point in time, was characterized by the control rod pattern, the amount of dissolved boron and the reactor thermal power. Nominal data were used throughout for the coolant flow rate and the core inlet temperature.

In-core instrumentation recordings were used to check the adequacy of the power distribution as calculated by PRESTO. The program includes a method of comparison with in-core, moveable, fission chamber detectors. Results of measured and calculated fission chamber traces from mid Cycle 1 are shown in Figure 7-11.

The reactor is equipped with a system of fixed, in-core detectors for power distribution monitoring. The process computer program INCA is used to infer power distributions from the instrument readings. Comparisons between INCA and PRESTO results are shown in Figure 7-12 for BOC1, in Figure 7-13 for EOC1, and in Figure 7-14 for BOC1A. In general, the agreement between calculations and measurements is satisfactory. This is particularly true for the assemblies where the instruments are placed. If the normalization is limited to these assemblies, the difference is within 3-4% in Cycle 1 and 7-8% in Cycle 1A. The deterioration in the agreement for Cycle 1A is mainly caused by assymetries in the fuel loading, resulting from the replacement of fuel assemblies before Cycle 1A. The INCA results which are obtained by mapping the instrumented readings over the core into one octant, are directly affected by these non-symmetry conditions.

The operating history was divided into 15 timesteps for Cycle 1 and 7 timesteps for Cycle 1A. For each timestep, the reactor was assumed to have been operated under constant conditions, as given in Table 7-6. The resulting reactivity curve is shown in Figure 7-15. The saw-tooth shape of the curve is due to the application of a fixed, soluble boron content within each timestep.

The minimum in the reactivity curve, at mid Cycle 1, is probably due to a slightly too low rate of shim rod depletion in the calculation. Model improvement may be desired on this point, from a reactor physics viewpoint. For the fuel duty analysis, however, this reactivity deficiency is of little concern. Core power distributions were generated and stored in the data bank at the end of each timestep.

Duty Cycle Simulation

The fuel duty cycle analysis, by means of the FDCA1 program, consists of combining the nodal power distributions from PRESTO with pin-group power distributions from RECORD and the approximated core power history shown in Figure 4-3. The fuel duty cycle is described by a sequence of events that may have produced local or global power shocks. The following were considered as events:

- Start-up with a power distribution different from that existing prior to shutdown.
- Rod movements at power.
- Power distribution shifts due to xenon-transients.
- Total power increase

Twenty events in Cycle 1 and 8 events in Cycle 1A were selected for this study. A list of the relevant data describing these events is given in Tables 7-11 and 7-12. The assembly power distribution after selected events, the average axial power distribution, the power shock matrix and the associated failure distribution are given in Appendix C.

A control-rod-induced Xe-oscillation event, of 26 November 1973, was analyzed in detail, using the Xe-dynamics option of the simulator. Another Xe-induced event, of 12 March 1974, was also analyzed in detail. The associated power shocks were not very marked (see Table 7-13). No significant power shocks were generated as a direct result of control rod motions, except for one event, (No. 7, Table 7-11), where the reactor was operated with Rod Group 5 withdrawn after a long period of operation with that group inserted. Rod Group 4 was inserted and withdrawn twice (Event Nos. 12 and 19), during the latter part of Cycle 1.

The remaining events analyzed for Cycles 1 and 1A were caused by changes in the total core power level.

FAILURE PREDICTIONS AND FISSION PRODUCT RELEASE

Failure Predictions

The nodal pin-group power density and power shock histories, as generated by the FDCA1 program, were saved on a permanent data file, called the Q-file, for subsequent analyses by the FDCA2-POSHO program.

The latter program was used to generate the expected number of cracks, failed pins and failed assemblies, including the locations of the failures, for each event analyzed and accumulated over each operating cycle. The number of cracks, failed pins and failed assemblies per event, together with the maximum power shock observed for each event, may be found in Tables 7-13 and 7-14. The end of cycle assembly failure probability distributions in a core octant for Cycles 1 and 1A, respectively, are given in Figures 7-16 and 7-17. Also given, are the corresponding "observed failure frequency" distributions, inferred from the sipping results shown in Figures 7-7 and 7-8.

The observed failure frequency for a given octant core position is here defined as the percent failed assemblies out of the total number (maximum 8) of assemblies, located in symmetric positions over the whole core.

Table 7-15 gives the predicted number of failed assemblies of each fuel type and the total number for the whole core, in comparison with sipping results for Cycle 1. Corresponding data for Cycle 1A are given in Table 7-16. Table 7-17 gives a comparison of prediction and sipping results for the sum of the Cycles 1 and 1A failures.

Cycle 1 measurements detected 43 leaking assemblies. Out of the 43 failed assemblies, 41 were of Type B, one of Type A and one of Type C. The corresponding predicted numbers were 2.7, 3.1 and 1.8 for Type A, B and C, respectively. Out of the 20 events analyzed for Cycle 1, none could be singled out as giving larger power shocks to Type B fuel than to Type A. There are, however, differences in the Types A and B assemblies as noted below.

	Type A	Type B
Poison Rods	No	Yes
Detector String	No	Yes
Enrichment (wt% U235)	2.01	2.40
Planned Residence in Core (Cycles)	1	2

As shown in Table 7-18, these parameters cause a general decrease (4%) in power for Type A but a general increase (3%) for Type B during the reactor cycle. (Combustion Engineering's calculations show a 6% rather than a 3-4% swing.) Having the detector string in the Type B central guide tube position will replace water, resulting in a lower center-of-assembly neutron flux than occurs for Type A. For an equivalent assembly power, this would mean that a slightly higher flux would exist at the four peripheral guide tube positions in the Type B assemblies. In the rod-by-rod examination performed as part of the EPRI contract RP586 Task C work scope (Reference 17), the preponderence of failed rods surrounded these higher power peripheral guide tube positions. The power sensitivity of the observed phenomenon is obviously very marked, as was analytically verified in EPRI contract RP397. (See Reference 17)

Figure A-3 of Reference 17 (EPRI NP-218) shows a perforated rod adjacent to the center guide tube location for Type A assembly A047. The post-irradiation investigation into the characteristics of this rod indicated an unusually high amount of fuel pellet in-reactor densification; this phenomenon increases the fuel-cladding gap and is, therefore, an alternative process (to higher linear power) for raising the fuel temperature and initiating the stress corrosion failure process.

Accurate accounting for this failure-initiating temperature inversion requires code logic which predicts the detail of fission product generation and storage within the fuel pellets, fission product release, change in fuel-cladding gap dimension and heat transfer coefficient, and synergistically combines these details through appropriate fuel-temperature behavioral feedback mechanisms. This level of detail is not present in the POSHO model used in these studies, and the Type A - Type B differences which were factored into the analysis were not sufficient to give any discrimination between these fuel types.

It is evident from Figures 7-7 and 7-16 that the Type B failure frequency varies radially and circumferentially across the core. These assymmetries are quantified below; the values shown are taken directly from Figures 7-7 and 7-8.

Radial Zone	No. Type B	Failed Type B		Fraction Failed	
		Cycle 1	Cycle 1A	Cycle 1	Cycle 1A
1	13	4	2	.307	.500
2	15	11	0	.733	0
3	21.5	18.5	0	.860	0
4	30.5	7.5	5	.246	.164
	80.0	41.0	7		

Octant Zone	No. Type B	Failed Type B		Fraction Failed	
		Cycle 1	Cycle 1A	Cycle 1	Cycle 1A
1	10	5.5	1	.55	.10
2	↓	5.5	2.5	.55	.25
3	↓	6	0.5	.60	.05
4	↓	6.5	0.5	.65	.05
5	↓	5.5	0.5	.55	.05
6	↓	3.5	0	.35	0
7	↓	4	2	.40	.20
8	↓	4.5	0	.45	0
	80	41.0	7.0		

Radial zone number one is at the center of the core, and the octant zones are counted in a counterclockwise direction starting from core position Y11.

The regions of highest failure frequency for Type B fuel generally coincide with the regions of highest radial power, especially when Rod Group 5 is inserted, as it was for a substantial period of Cycle 1. In addition, the radial regions of highest power will also experience the highest amplitude during Xe-induced oscillations of the axial power distribution.

Examples of failure probabilities for the xenon events analyzed are shown in Appendix D, Event Nos. 10 and 16. The 1/4-assembly failure probability distribution for Event No. 10 represents a typical xenon oscillation of the axial power distribution, following a relatively short period of reduced power operation. Event No. 15 represents a typical xenon oscillation following a reactor startup after a short shutdown period. The failure distribution in these events, although low, shows a tendency to favor of ring of assemblies in a radial mid sector of the core, in qualitative agreement with the radial profile noted above. The xenon-induced power shock distributions resemble the distribution of failed assemblies in Cycle 1.

The ability of the model to predict when the failures occur is illustrated in Figure 7-18 for Cycle 1, and Figure 7-19 for Cycle 1A. The calculated I^{131} activity level is here compared to the measured level in the coolant versus time. Note that the activity level is plotted on a logarithmic scale.

Fission Product Release Calculations

Calculations of the primary coolant I^{131} activity were performed using the model described in Section 4. The primary coolant purification system was operated at a flow-rate of 110 gpm up to January 1974 when it was increased to 140 gpm for the rest of the cycle. During Cycle 1A, the flowrate was 110 gpm. This has been accounted for in the calculation. The power reduction in December 1974 has been used to calibrate the exponent (m) in the escape rate coefficient correlation, yielding a value of approximately 1.5. This value has been used both for Cycle 1 and 1A. There is a large discrepancy in the actual versus predicted number of failed B assemblies (Cycle 1 and 1A) and, therefore, the calculated coolant activity level cannot be adjusted to the observed values. A figure of 5×10^{3} µCi/ml per failed pin at full power was used to scale the model at EOC1. The results of the calculations are compared with the observations in Figure 7-18 and Figure 7-19 for Cycles 1 and 1A.

EVALUATION OF RESULTS

The Power History Data Base

The procedure of the fuel duty cycle analysis for developing a detailed power history base relies on the ability of an analytic model to simulate core performance and the accuracy with which the operating history can be simulated. Comparisons between measured and calculated in-core signals and power levels in assemblies where in-core monitors were located, show that the PRESTO simulator adequately reproduces the power distributions in the core throughout the two cycles. Uncertainties introduced by the use of 1/8-core symmetry in Cycle 1A, where some assymetries in the loading existed, are not considered critical in this study.

The data base available at Maine Yankee for simulation of the operating history of the plant was considered quite satisfactory. The simulation of core burnup and the power shocks resulting from gross power changes and rod motion were readily accomplished. The banked operation of control rods and the lower frequency of rod motions make simulation of steady state power distributions considerably easier than in a BWR, such as Quad Cities Unit Two. On the other hand, simulation of the power shocks resulting from power changes during xenon transients assumes more importance in a PWR. Significant power shifts were found in the two xenon transients included in the analysis, but due to the relatively slow ramp rates in such transients, the consequences in fuel failures were calculated to be small. The power shocks developed in the events anlyzed at Maine Yankee are, in general, small, 25 - 50 w/cm, but involve a large part of the core. (See power shock matrix in Appendix D.) The maximum power shock occurring in the core is less than 100 w/cm. This is a typical result for a PWR and is considered to be the main reason why there are, normally, fewer PCI failures in a PWR. (Many shocks occurred in Quad Cities Unit Two of over 100 w/cm, with a maximum calculated shock of 350 w/cm.) Operation of a PWR with chemical shim control, rather than control rods, reduces the frequency of shocks which would result from control rod motion.

Failure Predictions

The total number of failed assemblies predicted by FDCA-POSHO over the two cycles is in good agreement with the sipping results for Assembly Types A and C. The prediction for Fuel Type B is nearly the same as for A, but the sipping results show that nearly 10 times as many B Assemblies actually failed. As discussed previously, fuel Types A and B are, with respect to the POSHO model, of identical design. Because the high failure rate for the Type B fuel cannot be described adequately by the normal PCI model of FDCA-POSHO, it is of interest to try to determine if these failures are caused by power shocks or not.

FDCA-POSHO over-predicts the failures in Fuel Types A and C during Cycle 1, but under-predicts the failures in Cycle 1A. For the two cycles together, the agreement is very good. This is the same trend as for Quad Cities 2, and can be rationalized by the same arguments.

The failure prediction level in FDCA-POSHO is expected to be slightly high, since POSHO has been adjusted to the failure data from a number of operating plants with an approximated representation of the power distribution history, which did not account for the large number of small power shocks found in the more detailed analyses performed for Maine Yankee. The over-prediction in Cycle 1 and under-prediction in Cycle 1A also might be explained if some failed assemblies were not detected by the sipping at EOC-1, but were detected at EOC-1A. The under-prediction in Cycle 1A may also be attributed to the fact that the POSHO model employed does not account for the cumulative effect of previous shocks on the fuel's resistance to PCI failures.

Comparisons between the accumulated number of predicted failed pins and the I^{131} coolant activity versus time is difficult, since the activity level is dominated by the high failure rates in the B assemblies. In Cycle 1, it is observed that the bulk of these assemblies apparently failed in January and March 1974 and during the power escalation to 95% power in the beginning of April. In spite of the above and of the gross under-prediction of failed pins, it is observed that the timing of predicted failures does coincide with increases in coolant activity.

It has been noted previously that the distribution of the failed B Assemblies at EOC-1 is not random, but concentrated around a mid-radial zone in the core. The accumulated failure predictions for Cycle 1 also are slightly higher in this region. Reviewing the individual events, it is observed that this is mainly caused by the two xenon events. If one assumes that these events are representative of similar transients not accounted for, it is evident that the resolution could be improved by including these. The predicted failure frequencies, even including such additional events, still would not approximate the actual failure frequencies. To achieve this, the failure model for the Type B fuel would have to have a threshold-type power level dependence.

The results of the joint EPRI/Combustion Engineering evaluation of fuel rod performance in Maine Yankee Core 1 has, subsequent to the completion of this analysis, been made available (17). Combustion Engineering attributes the failures in this fuel to a "thermal instability mechanism," involving:

> ".....densification which increased the pellet to clad gap, which increased the fuel temperature, which increased fission gas release, which reduced gap conductance, which increased fuel temperature, etc....."

The high fission gas level in the pins and localized stress concentration (PCI) are then believed to have caused stress corrosion cracking (SCC). The fission gas release mechanism is triggered at a certain threshold power level and, if a cladding stress is subsequently maintained, SCC failure results. If such a phenomenon really did exist, the xenon transients might have produced the crack-initiating power shocks and thus could explain the non-random distribution of the failed Type B fuel assemblies. Additional power shocks would increase the failure probability, but evidently were not necessary.

Considering the statistical variations in fuel parameters and material properties alone, a more random distribution of failed B Assemblies might be expected if the above was the cause of failure. However, the power level in the failed assemblies was only marginally (10%) higher than in the unfailed assemblies in the central region of the core. Thus, when combined with the power shocks which tended to produce a ring zone of higher failure probabilities in the locations of the actual failure locations, the deductions made from this study are in agreement with the above explanation.

Fission Product Release Algorithm

The fuel failure predictions for Maine Yankee do not correspond to the sipping results, because the model does not predict the high failure frequency of the B-type assemblies. The calculated coolant activity level, based on the predicted number of failures, is one twenty-fifth of the measured value. Using the observed activity level and the release algorithm one would calculate that approximately 200 pins failed in Cycle 1. This result would represent an average of more than 5 pins failed per failed B-type assembly. The model predicts not more than one pin failed per assembly. Post-irradiation examination of the failed A-assembly and C-assembly reveals one leaking rod in each. The two B-assemblies that were disassembled, however, had 4 and 11 failed pins (17).

It is seen from a comparison between the measured and the calculated activity level that the power history data base contains the major events both for Cycle 1 and 1A, and that the model calculates failures at these occurrences. The model can not predict the behavior of the Type B fuel. It can be seen, however, that some of the major events may have triggered the failure mechanism in these assemblies, as the large coolant activity increases coincide with these events.

TABLE 7-1

GEOMETRIC DESIGN DATA FOR INITIAL AND RELOAD FUEL. ALL DIMENSIONS ARE IN CM.

MAINE YANKEE

Fuel pellet radius	0.4820
Fuel pellet length	1.651
Dish radius	0.37
Pellet-clad gap (radial)	0.0108
Clad, inner radius	0.4928
Clad, outer radius	0.5588
Clad thickness	0.0660
Fuel rod pitch	1.4732
Fuel rod array	14 x 14
Assembly pitch, fuel rod to fuel rod	20.2692
Thickness of water gap	0.1524

TABLE 7-2

FUEL AND ASSEMBLY MATERIAL COMPOSITION FOR FUEL IN CYCLES 1 AND 1A

MAINE YANKEE

Description \ Fuel type	A	B	C1	C2	C3	RF0	RF4	RFB	A5
Weight of U pr. assembly (kg)	394.8	358.9	394.8	367.9	358.9	394.8	358.8	394.8	383.6
No. of assemblies (Cycle 1)	69	80	24	36	8	-	-	-	-
No. of assemblies (Cycle 1A)*	57	24	22(24)	34(36)	8	14(12)	55(56)	2(0)	1(0)
No. of fuel rods/assembly	176	160	176	164	160	176	172	176	171
Fuel enrichment (wt% U235)	2.01	2.40	2.95	2.95	2.95	1.93	1.93	2.33	2.01
Uranium in fuel (wt%)	0.8815	0.8815	0.8815	0.8815	0.8815	0.8815	0.8815	0.8815	0.8815
Fuel (UO_2) density (g/cm^3)	10.193	10.193	10.193	10.193	10.193	10.193	10.193	10.193	10.193
No. of shim rods	0	16	0	12	16	0	4	0	5
Clad	Zircaloy-4								
Clad density (g/m^3)	6.55								

* The numbers in parenthesis are those used in the simulation of Cycle 1A, assuming octant symmetry.

TABLE 7-3

MAINE YANKEE SHIM ROD DATA (See Figs. 7-3 & 7-4)

Active length	3.1166 m
Material	B_4C-Al_2O_3
Pellet Diameter	0.96266 cm
Clad Material	Zircaloy-4
Clad Inner Diameter	.98552 cm
Clad Outer Diameter	1.1176 cm
Wt% B_4C in Al_2O_3 (90% theoretical density):	
Fuel Type B	1.95
Fuel Type C	0.75
Fuel Type RF	2.83

TABLE 7-4

CONTROL ELEMENT ASSEMBLY DATA (SEE FIG. 7-2).
ALL DIMENSIONS ARE IN CM.

	Full length	Part length
Number	77	8
Number of absorber elements	5	5
Type	Cylindrical	Cylindrical
Radius of element	1.20396	1.20396
Clad material	Inconel 625	Inconel 625
Clad thickness	11.7856	11.7856
Total element length	409.734	409.734

TABLE 7-5

MAINE YANKEE CORE DESIGN DATA

Number of fuel rods (Core 1)	36352
Total number of fuel assemblies	217
Number of control element assemblies (CEA)	85
CEA pitch (cm)	29.39
Hydraulic diameter, nominal channel (cm)	1.3548
Total flow area (m^2)	4.9703
Total heat transfer area (m^2)	4431.473
Heat transfer area for assembly (m^2)	20.4215
Core height (active fuel, cm)	347.218

TABLE 7-6

MAINE YANKEE NOMINAL OPERATING DATA

Core thermal power (MW)	2440
Pressure (psig)	2235
Coolant inlet temperature (^{o}C)	281.6
Core bulk outlet temperature (^{o}C)	311.1
Coolant flow through core (kg/sec)	$1.5322 \cdot 10^4$

TABLE 7-7

MAINE YANKEE OPERATING CONDITION DATA - CYCLE 1

STEP NO.	END OF STEP		STEP LENGTH	ROD GROUP POSITION		SOL. BORON	TH. POWER
	DATE	E(MWD/TU)	ΔE (MWD/TU)	GR. 4	GR. 5	ppm	(MW)
1	1/1-73	759	759	180	180	723	1830
2	1/3-73	1645	886	180	165	751	1830
3	1/4-73	2073	428	139	149	760	1830
4	1/6-73	3335	1262	165	3	725	1821
5	1/7-73	3804	469	160	3	750	1443
6	1/9-73	4121	317	6	3	735	1206
7	12/10-73	4573	452	179	160	748	1873
8	31/11-73	5650	1077	180	20	673	1958
9	31/12-73	6277	627	180	25	660	1964
10	31/1-74	7004	727	166	30	642	1858
11	2/3-74	7786	782	169	10	578	1943
12	5/4-74	8433	647	169	10	528	2193
13	30/4-74	8988	555	178	8	523	1975
14	21/5-74	9485	497	178	8	485	1989
15	28/6-74	10367	882	178	8	432	1960

TABLE 7-8

MAINE YANKEE OPERATING CONDITION DATA, CYCLE 1A

STEP NO.	END OF STEP		STEP LENGTH	ROD GROUP POSITION		SOL. BORON	TH. POWER
	DATE	E(MWD/TU)	ΔE (MWD/TU)	GR. 4	GR. 5	ppm	(MW)
1	7/11-74	567	567	180	161	519	1906
2	8/12-74	1224	657	180	164	515	1912
3	1/1-75	1805	581	180	164	450	2281
4	1/2-75	2337	532	180	164	452	1937
5	1/3-75	2956	619	180	164	417	2051
6	1/4-75	3740	784	180	166	363	2199
7	3/5-75	4500	760	180	166	313	2127

TABLE 7-9

NUMBER OF FUEL PINS PER PIN GROUP IN A 1/4-ASSEMBLY FOR EACH ASSEMBLY TYPE IN MAINE YANKEE

Assembly Type	No. of Pins per Group		
	1	2	3
A	11	23	10
B	11	19	10
C1	11	23	10
C2	11	20	10
C3	11	19	10
RF0	11	23	10
RF4	11	22	10

TABLE 7-10

PIN GROUP POWER (W/CM) FOR EACH FUEL TYPE AND FOR THE CONTROL ELEMENT IN/OUT CONDITIONS. MAINE YANKEE

Fuel Type	Control Element	Pin Group No.		
		1	2	3
A	Out	182.2	191.1	209.7
	In	210.1	193.7	173.9
B	Out	186.4	189.9	207.3
	In	214.9	192.2	171.9
C_1	Out	179.8	190.5	214.6
	In	207.3	192.8	177.9
C_2	Out	178.8	190.5	215.0
	In	206.2	192.8	178.2
C_3	Out	180.8	189.1	215.0
	In	208.5	191.4	178.3
RF_0	Out	183.19	191.76	207.97
	In	211.22	194.06	172.41
RF_4	Out	185.12	191.05	207.31
	In	213.44	193.34	171.86

TABLE 7-11

MAINE YANKEE DESCRIPTION OF EVENTS - CYCLE 1

Event No.	Date of Event	POWER Level (%) Before Ramp	After Ramp	Control Group Positions Bank 4	Bank 5	Ramp Time (Hrs.)	Comments
0	BOC	0	75	180	155	120	Startup Cy 1
1	2/7 - 73	50	75	180	160	24	Power increase
2	3/14	0	75	170	160	24	Fast startup
3	4/5	75	75	160	0	0	Bank 5 inserted
4	5/5	0	75	155		24	Fast startup
5	6/10	33	70	160		96	Power variation
6	8/5	0	50	0	0	72	Low power operation
7	10/1	0	81.8	180	160	96	Startup. Bank 5 withdrawn
8	10/27	81.8	73	180	25	72	Power reduction
9	11/26	0	77	180	25	24	Fast startup
10	11/27 - 12/29	61	78	Variable	Variable	3	Xenon transient
11	1/1 - 74	78	77	180	25	72	Steady power operation
12	1/14- 74	0	84		30	24	Fast startup
13	1/14 - 3/7	10	87	180	30	48	Startup from 10% power
14	3/11	87	79	170	10	72	Steady power operation
15	3/11 - 31	0	83			26	Startup & xenon transient
16	4/5	83	83			33	Steady operation
17	4/11	83	91.7	170		96	Power increase
18	4/11 - 5/21	0	92	180		72	Startup
19	5/23	92	78			72	Steady power operation
20		48.5	82	180	10	26	Steady operation

Table 7-12

MAINE YANKEE Description of Events - Cycle 1A

Event No.	Date of Event	POWER Level (%) Before Ramp	POWER Level (%) After Ramp	Control Group Positions Bank 4	Control Group Positions Bank 5	Ramp Time (Hrs.)	Comments
1	10/10 - 74	0	76	180	160	168	Startup Cy-1A
2	11/16	0	76		160	48	Startup
3	12/3	76	86		165	120	Power increase to 86%
4	12/9	0	90.3			8	Startup to 90.3%
5	12/20-2/28-75	90.3	81			24	Steady operation at 81%
6	2/28	0	81			34	Startup
7	3/14	73	87			120	Power increase to 87%
8	4/14 - 5/2	87	78	180	165	48	Steady operation at 78%

Table 7-13

Maximum power shocks and predicted failures per event - Cycle 1

MAINE YANKEE

Event No.	ΔQmax (w/cm)	Octant Failure Prediction Assemblies	Pins	Cracks
0	0.	0.	0.	0.
1	25.	0.	0.	0.
2	25.	0.	0.	0.
3	25.	0.	0.	0.
4	25.	0.	0.	0.
5	25.	0.	0.	0.
6	0.	0.	0.	0.
7	100.	0.2	0.2	0.2
8	25.	0.	0.	0.
9	75.	0.1	0.1	0.1
10	50.	0.1	0.1	0.1
11	25.	0.	0.	0.
12	75.	0.2	0.2	0.2
13	50.	0.1	0.1	0.1
14	25.	0.	0.	0.
15	50.	0.1	0.1	0.1
16	25.	0.	0.	0.
17	25.	0.1	0.1	0.1
18	50.	0.1	0.1	0.1
19	0.	0.	0.	0.
20	25.	0.	0.	0.

TABLE 7-14

MAXIMUM POWER SHOCKS AND PREDICTED FAILURES PER EVENT - CYCLE 1A

MAINE YANKEE

Event No.	ΔQ_{max} (w/cm)	Octant Failure Prediction Assemblies	Pins	Cracks
1	100.	0.2	0.2	0.2
2	50.	0.	0.	0.
3	25.	0.1	0.1	0.1
4	50.	0.2	0.2	0.2
5	25.	0.	0.	0.
6	25.	0.	0.	0.
7	25.	0.1	0.1	0.1
8	0.	0.	0.	0.

Table 7-15

Actual and Predicted Failures by Assembly Type - Cycle 1

MAINE YANKEE

Assembly Type	A	B	C	Total
No. in core	69	80	68	217
No. of failed assemblies	1	41	1	43
No. of failed assemblies predicted	2.7	3.1	1.8	7.6
% Failed	1.4	51	1.5	19.8
% Failed predicted	3.8	3.9	2.7	3.5

Table 7-16

Actual and Predicted Failures by Assembly Type - Cycle 1A

MAINE YANKEE

Assembly Type	A	B	C	RF	Total
No. in core	57	24	64	72	217
No. of failed assemblies	3	7	2	0	12
No. of failed assemblies predicted	2.5	.5	0.7	0.0	3.7
% Failed	5.3	29.2	3.1	0	5.5
% Failed predicted	4.4	2.0	1.1	0.0	1.7

TABLE 7-17

ACTUAL AND PREDICTED FAILURES, SUM OF CYCLES 1 AND 1A

MAINE YANKEE

Assembly Type	A	B	C	RF
Batch Size	69	80	68	72
No. Failed in Cycles 1 and 1A	4	48	3	0
Prediction Sum of Cycles 1 and 1A	5.2	3.6	2.5	0.0

TABLE 7-18

CORE AVERAGE RELATIVE PIN POWERS AND BURNUP PER ASSEMBLY TYPE FOR CYCLE 1 AND 1A (FROM PRESTO)

MAINE YANKEE

Assy. Type	Relative Power				Burnup (MDW/TU)	
	BOC-1	EOC-1	BOC-1A	EOC-1A	EOC-1	EOC-1A
A	1.095	1.053	1.032	1.010	11075	15830
B	1.065	1.093	1.055	1.078	11170	15151
C	0.832	0.845	0.759	.785	8778	12108
RF			1.190	1.181		5351

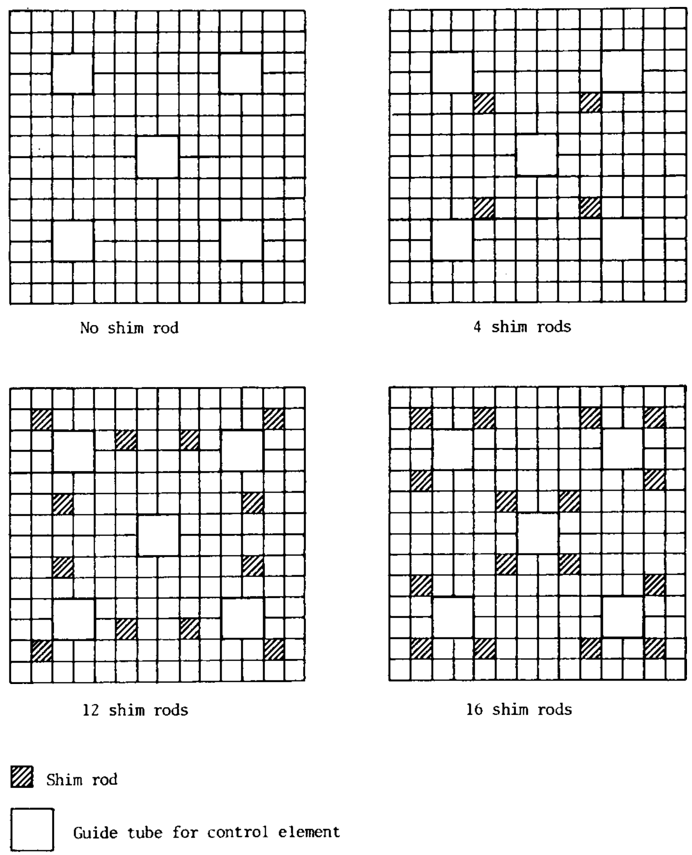
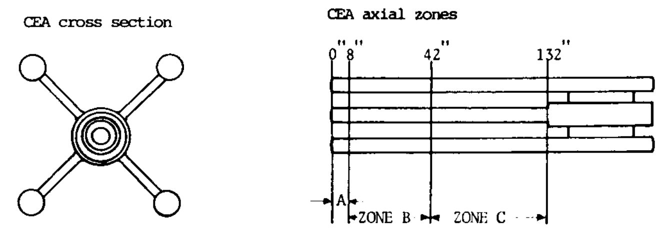


FIG. 7.1 FUEL ASSEMBLY LAYOUT FOR EACH TYPE OF SHIM ROD ARRANGEMENT.

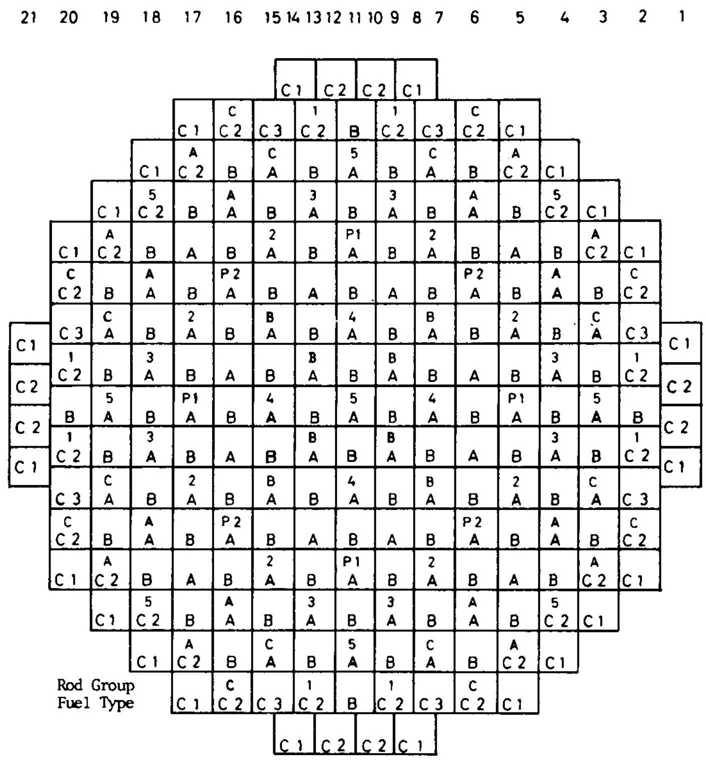


Rod no.	Number	Core Location	Materials* ZONE A	B	C
Full length rods					
1	69	All others	Ag Ag / B / Ag Ag	B B / B / B B	B B / B / B B
2	4	D-4	S S / B / S S	S S / B / S S	S S / B / S S
3	4	C-11 L-3	Ag S / S / S Ag	B S / S / S B	B S / S / S B
Part length rods					
P-1	4	L-11 L-5	S S / S / S S	B B / B / B B	Al Al / Al / Al Al
P-2	4	F-6	S S / S / S S	S S / B / S S	Al Al / Al / Al Al

* Ag = Ag-In-Cd S = Stainless Steel
B = B_4C Al = Al_2O_3

FIG. 7.2 CONTROL ELEMENT ASSEMBLY DESIGN.

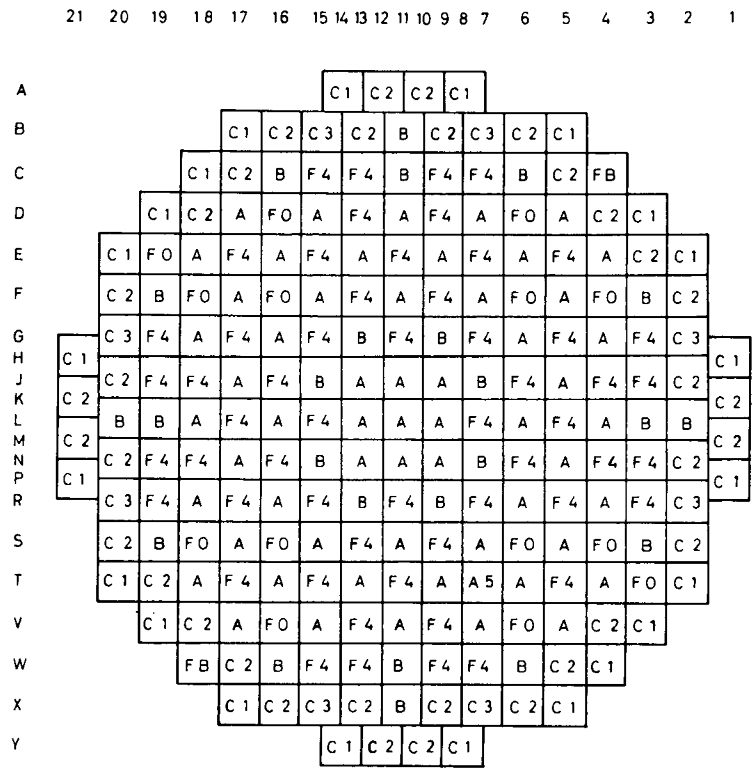
MAINE YANKEE



A	2.01 wt%	No Shim rods	Rod gr. A -	Rod Type 1
B	2.40 "	16 " "	" " B -	" " 1
C1	2.95 "	No " "	" " C -	" " 1
C2	2.95 "	12 " "	" " 1 -	" " 1
C3	2.95 "	16 " "	" " 2 -	" " 1
			" " 3 -	" " 1
			" " 4 -	" " 1
			" " 5 -	" " 1,2,3
			" " P-1 -	" " P-1
			" " P-2 -	" " P-2

FIG. 7.3 CORE ARRANGEMENT - CYCLE 1

MAINE YANKEE



A -	Partially burned		2.01 wt%	No shim rods
B -	"	"	2.40 "	16 " "
C1-	"	"	2.95 "	No " "
C2-	"	"	2.95 "	12 " "
C3-	"	"	2.95 "	16 " "
F0(RF0)-	Fresh		1.93 "	No " "
F4(RF4)-	"		1.93 "	4 " "
FB(RFB)-	"		2.33 "	No " "
A5-	"		2.01 "	5 " "

FIG. 7.4 CORE ARRANGEMENT - CYCLE 1A

MAINE YANKEE

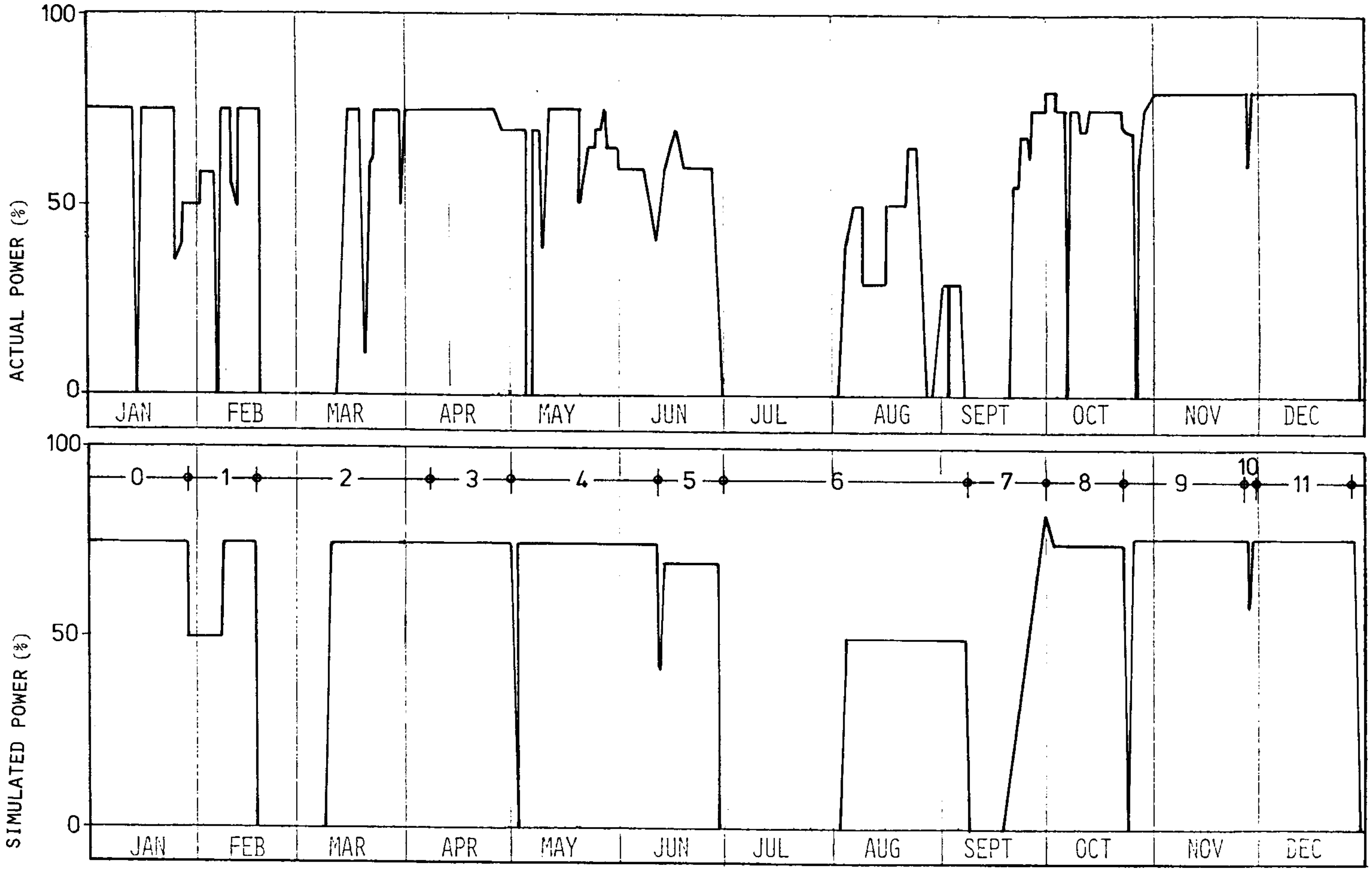


FIG. 7.5 ACTUAL AND SIMULATED REACTOR POWER HISTORY - CYCLE 1 1973
MAINE YANKEE

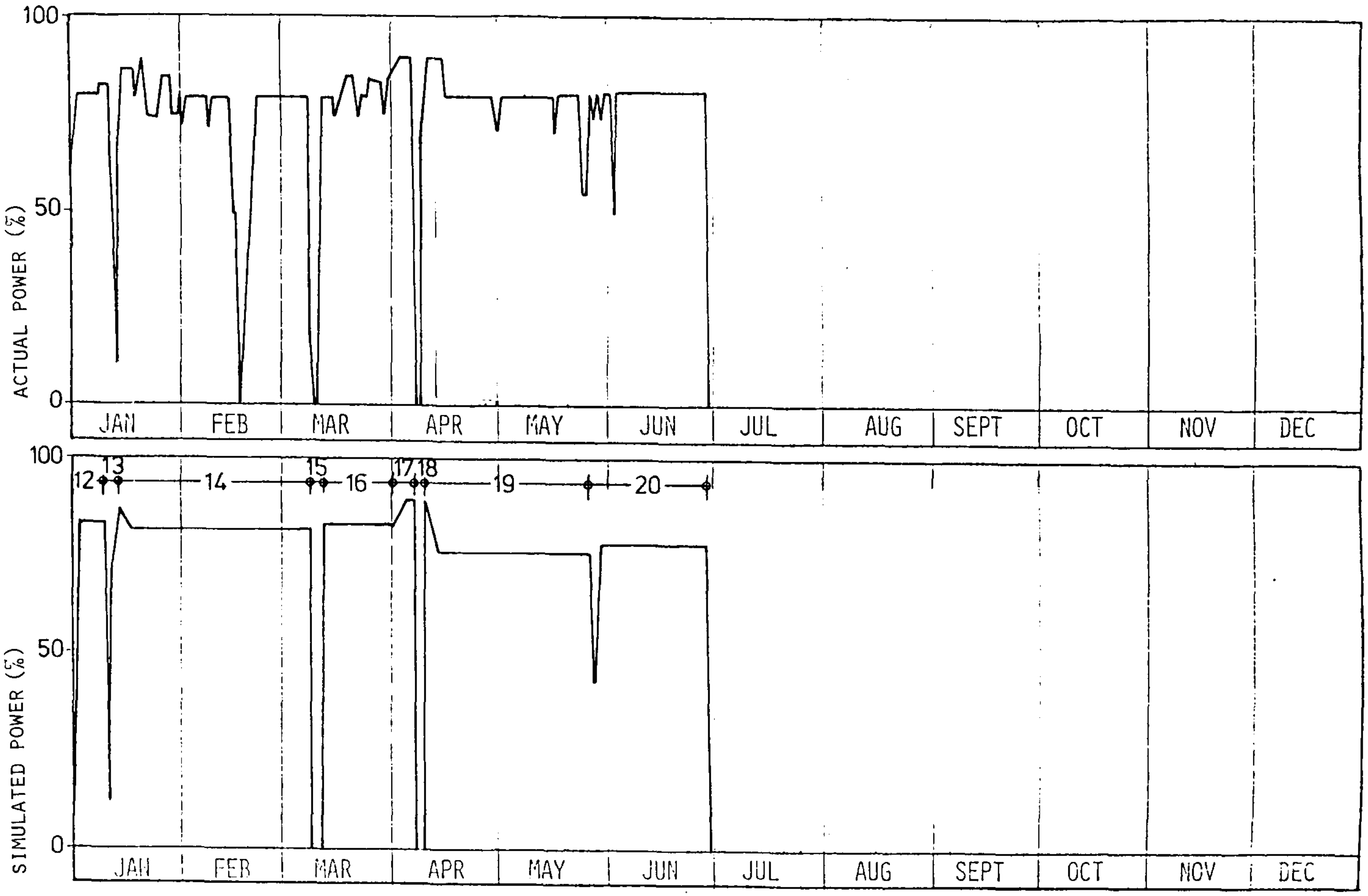


FIG. 7.5 ACTUAL AND SIMULATED REACTOR POWER HISTORY - CYCLE 1, CONTINUED, MAINE YANKEE

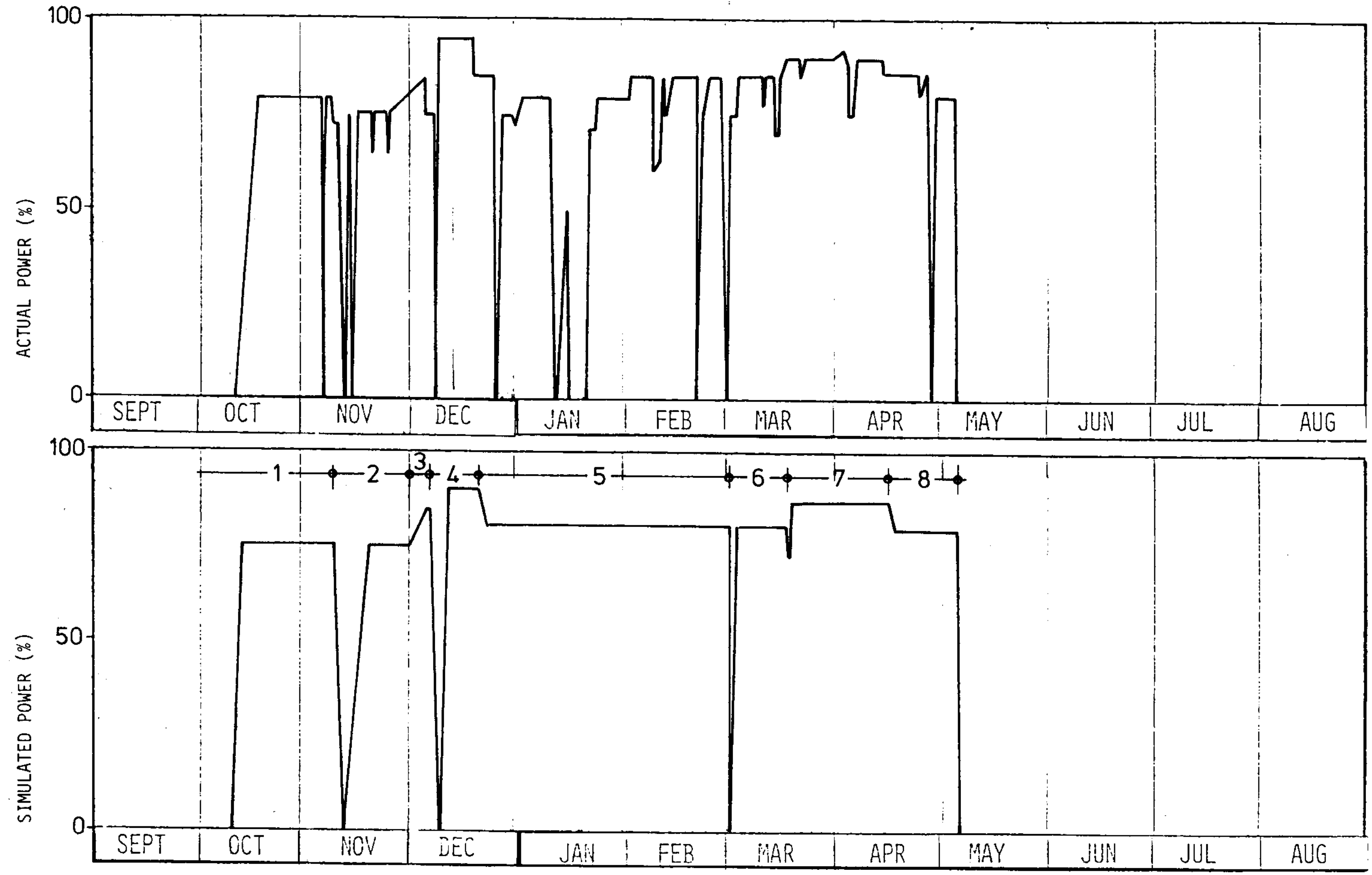
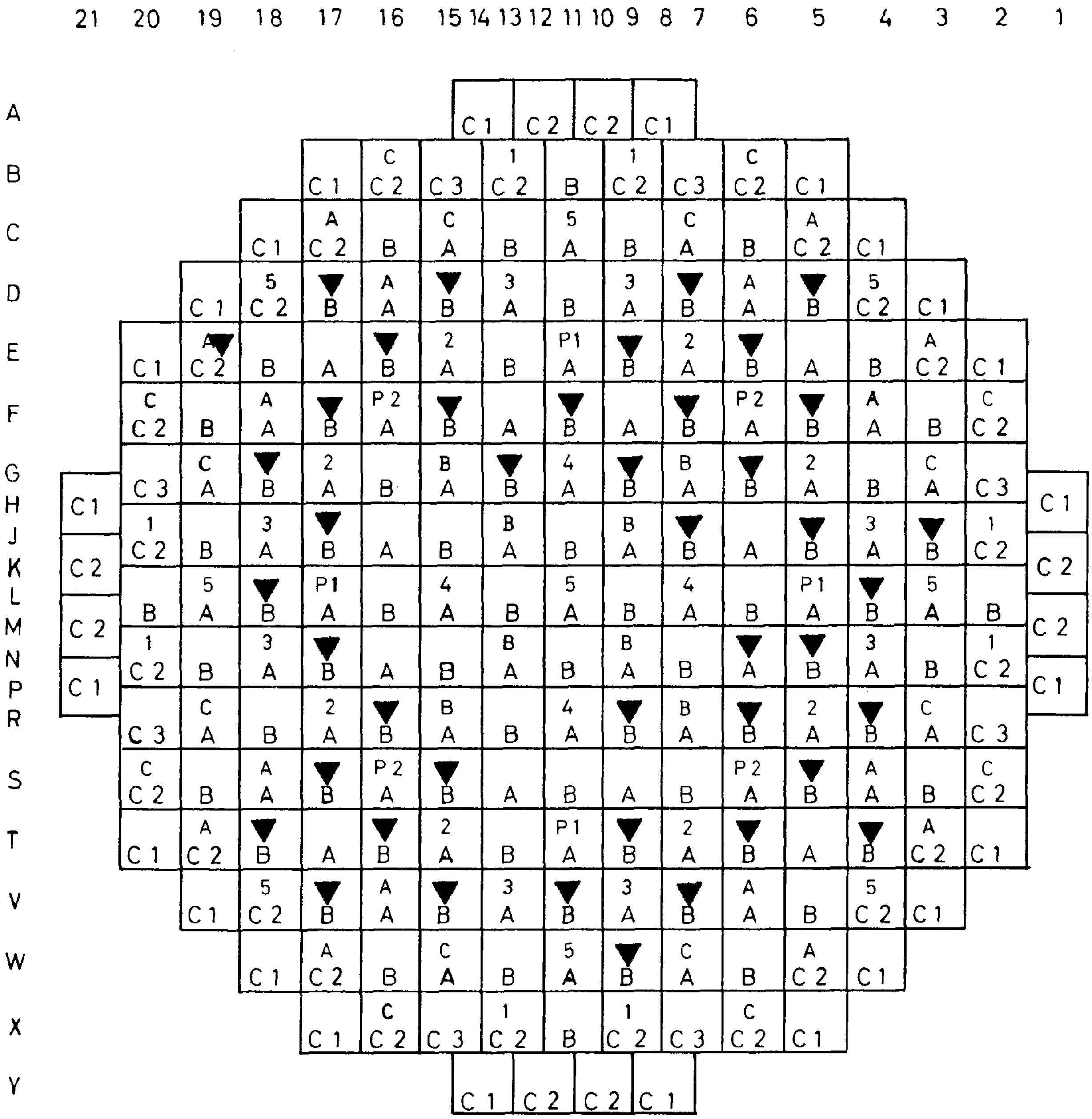


FIG. 7.6 ACTUAL AND SIMULATED REACTOR POWER HISTORY - CYCLE 1A 1974/75
MAINE YANKEE



▼ — Assembly contained failed fuel pins(s)

FIG. 7.7 LOCATIONS OF FAILED ASSEMBLIES AS IDENTIFIED BY SIPPING MEASUREMENTS - CYCLE 1. MAINE YANKEE

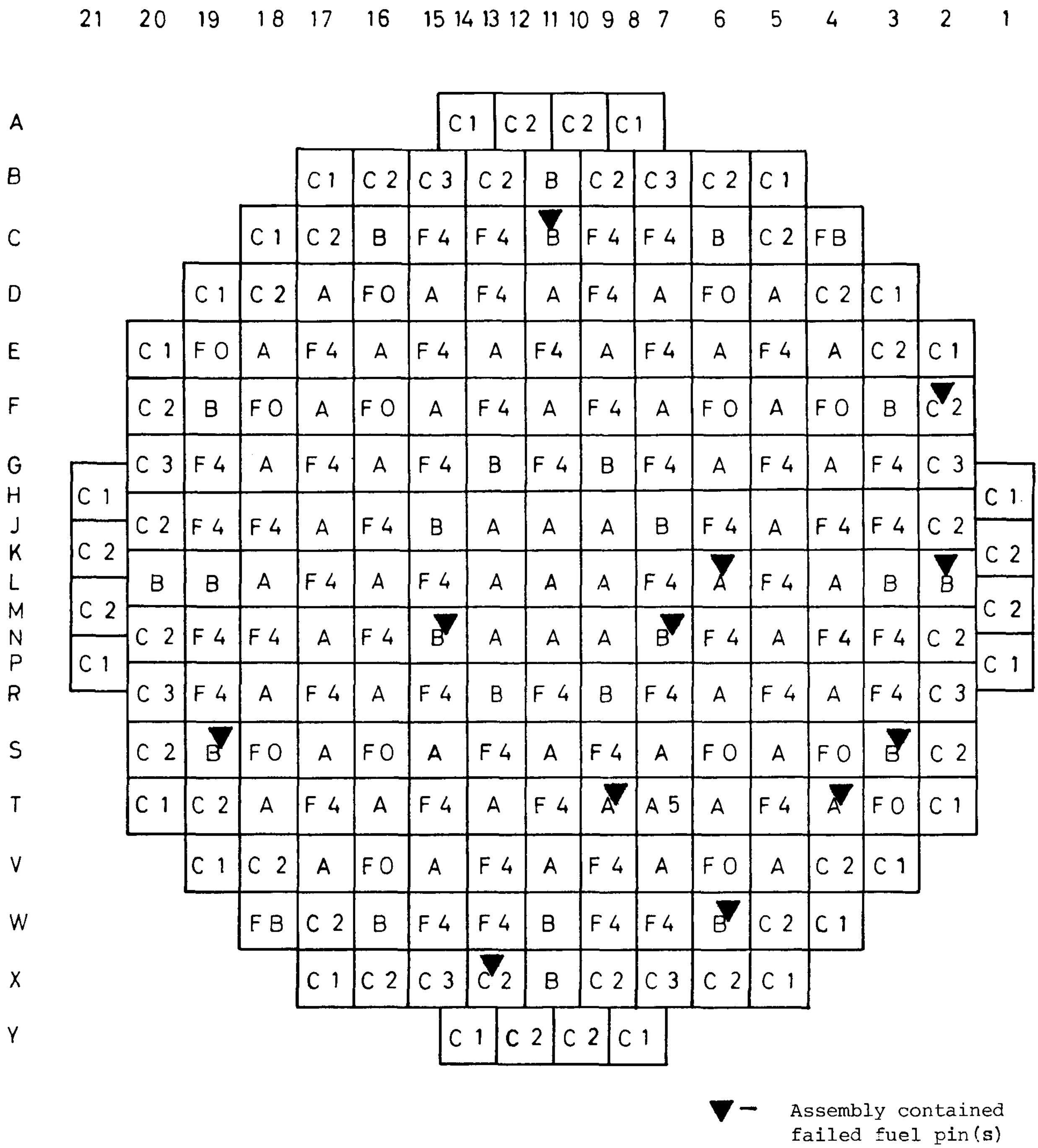


FIG. 7.8 LOCATIONS OF FAILED ASSEMBLIES AS IDENTIFIED BY SIPPING MEASUREMENTS - CYCLE 1A. MAINE YANKEE

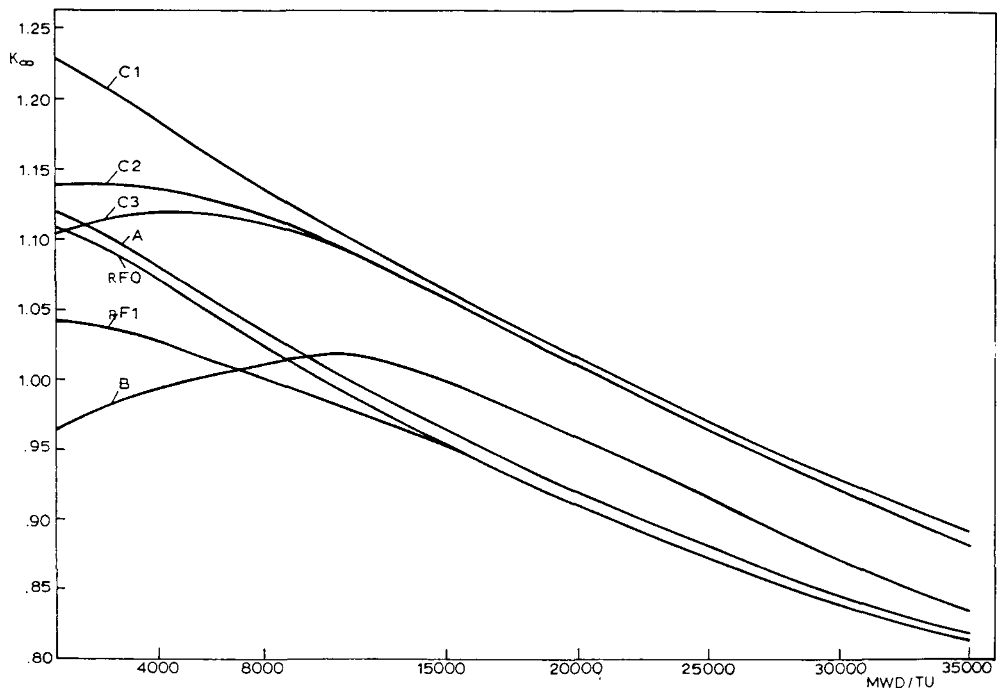


FIG. 7.9 K_{∞} VERSUS BURNUP AT 400 PPM SOLUBLE BORON CONTENT FOR ALL FUEL TYPES ANALYZED.

MAINE YANKEE

```
**RECORD 76-1**   OUTPUT    RE-MA   5  OFUEL TYPE A. NO CONTROL. BORON = 400

AVERAGE BURNUP =  4000. MWD/TU                    29.923 W/GM. VOID 0.000, FP POIS.. RCC OUT
                  -------------                   SOLB. POIS.   400.00 PPM

POWER MAP       POWER PEAKING FACTOR = 1.0863
---------
                MAXIMUM LINEAR LOADING IN FUEL ASSEMBLY = 210.00 WATTS/CM AT PIN NO. 10

POWER PER PIN IN 1/4 ASSEMBLY (RELATIVE TO UNIT AVERAGE POWER IN WHOLE ASSEMBLY)
```

1 2 1 ← Pin group no.

1 1.0037	2 1.0028	3 1.0170	4 1.0119	5 .9842	6 .9554	7 .9400
8 1.0028	9 1.0264	10 1.0863	11 1.0815	12 1.0083	13 .9504	14 .9261
15 1.0170	16 1.0863	17 *******	18 *******	19 1.0709	20 .9670	21 .9327
22 1.0119	23 1.0815	24 *******	25 *******	26 1.0737	27 .9757	28 .9461
29 .9841	30 1.0083	31 1.0709	32 1.0737	33 1.0160	34 .9821	35 .9793
36 .9553	37 .9504	38 .9670	39 .9757	40 .9821	41 1.0175	42 1.0768
43 .9400	44 .9261	45 .9327	46 .9461	47 .9793	48 1.0768	49 *******

1 3

******* DENOTES WATER HOLE

FIG. 7.10 EXAMPLE OF 1/4 ASSEMBLY PIN POWER DISTRIBUTION. PARTITION INTO PIN GROUPS IS SHOWN.

MAINE YANKEE

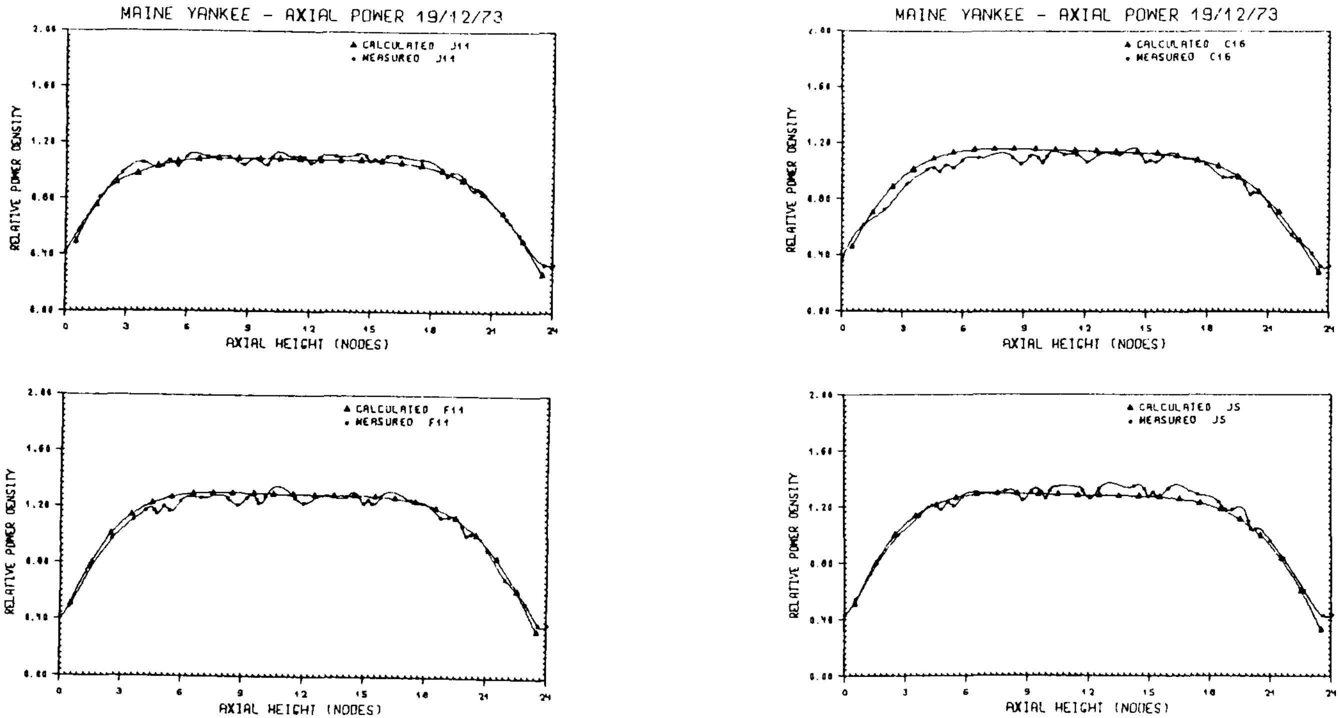


FIG. 7.11 EVALUATION OF CALCULATED POWER DISTRIBUTIONS COMPARISON WITH MOVABLE FISSION CHAMBER TRACES MID CYCLE-1.

INCA	1.22	1.11*	1.21	1.10*	1.19	1.08*	1.10	0.88*	0.68
PRESTO	1.22	1.10	1.20	1.08	1.18	1.04	1.12	0.89	0.71
		1.20	1.08*	1.17	1.06*	1.15	.98*	1.09	.61*
		1.22	1.09	1.19	1.06	1.14	1.00	1.13	.63
			1.17	1.06*	1.17	1.05*	1.08	.94*	
			1.20	1.07	1.16	1.02	1.08	.94	
				1.18	1.09*	1.12	.90*	.73	
				1.16	1.03	1.09	.88	.74	
					1.15	.99*	1.00	.58*	
					1.11	.95	1.02	.60	
						1.07	.74*		
						1.08	.74		

* INSTRUMENTED POSITION.

FIG. 7.12 EVALUATION OF CALCULATED POWER DISTRIBUTIONS. COMPARISON WITH INCA (IN-CORE) RESULTS BOC-1.

MAINE YANKEE

INCA PRESTO	.69 .73	1.02* .96	1.08 1.13	1.11* 1.09	1.12 1.14	1.08* 1.02	.86 .89	.96* .95	.74 .78
		1.04 1.09	1.08* 1.08	1.09 1.15	1.12* 1.09	1.10 1.11	1.07* 1.02	1.13 1.15	.64* .63
			1.08 1.15	1.10* 1.10	1.12 1.15	1.12* 1.08	1.08 1.09	.98* .99	
				1.12 1.15	1.15* 1.09	1.11 1.12	1.03* 1.00	.80 .83	
					1.12 1.11	1.07* 1.00	1.02 1.05	.60* .62	
						.86 .90	.67* .64		

* INSTRUMENTED POSITION.

FIG. 7.13 EVALUATION OF CALCULATED POWER DISTRIBUTIONS. COMPARISON WITH INCA (IN-CORE) RESULTS EOC-1.

MAINE YANKEE

INCA	.97	1.01*	1.10	1.11*	1.10	1.13*	1.12	.96*	.70
PRESTO	.86	.91	1.14	1.06	1.18	1.09	1.06	.90	.63
		1.08	1.08*	1.10	1.09*	1.14	1.18*	1.09	.57*
		.96	1.04	1.19	1.09	1.21	1.19	1.02	.51
			1.10	1.09*	1.13	1.17*	1.11	.96*	
			1.21	1.12	1.22	1.12	1.15	.88	
				1.11	1.18*	1.08	1.03*	.79	
				1.34	1.14	1.29	.96	.74	
					1.08	.95*	.94	.55*	
					1.17	.99	.93	.53	
						.98	.65*		
						.96	.59		

* Instrumented position.

Fig. 7.14 Evaluation of calculated power distributions. Comparison with INCA (in-core) results BOC-1A
MAINE YANKEE

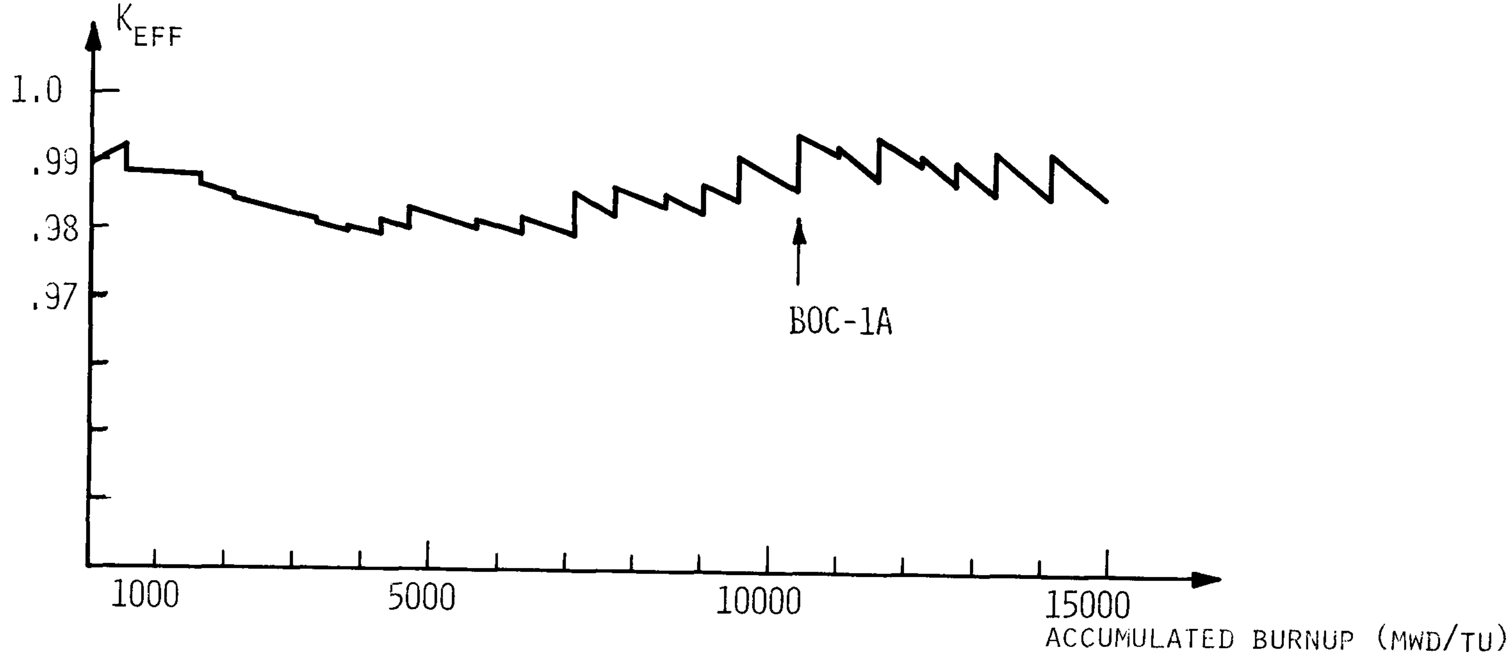


FIG. 7.15 CALCULATED REACTIVITY THROUGH CYCLES 1 AND 1A

MAINE YANKEE

A 3.2 / 0.0	B 3.2 / 0.0	A 3.2 / 0.0	B 3.6 / 25.0	A 3.8 / 0.0	B 4.2 / 75.0	A 4.8 / 0.0	B 3.6 / 0.0	C 1.6 / 0.0
	A 3.2 / 0.0	B 3.4 / 50.0	A 3.6 / 12.5	B 4.0 / 75.	A 4.0 / 0.0	B 4.4 / 0.25	C 6.2 / 0.0	C 0.8 / 0.0
		A 3.6 / 0.0	B 4.0 / 75.0	A 4.0 / 0.0	B 4.3 / 75.0	A 3.9 / 0.0	C 3.8 / 0.0	
			A 4.0 / 0.0	B 4.4 / 100.	A 4.0 / 0.0	B 3.3 / 0.0	C 1.9 / 0.0	
				A 4.0 / 0.0	B 3.6 / 62.5	C 4.2 / 12.5	C 0.6 / 0.0	
					C 5.1 / 0.0	C 0.9 / 0.0		

← CALC. FAILURE PROB. (%)

← OBSERVED FAILURE FREQUENCY (%)

ASSEMBLY TYPE →

FIG. 7.16 COMPARISON OF CALCULATED ASSEMBLY FAILURE PROBABILITIES WITH OBSERVED FAILURE FREQUENCY DISTRIBUTION OVER A CORE OCTANT - CYCLE 1.

MAINE YANKEE

A 1.2 / 0.0	A 1.4 / 0.0	F 0.0 / 0.0	A 2.0 / 25.0	F 0.0 / 0.0	A 2.0 / 0.0	B 2.8 / 25.0	B 1.8 / 25.0	C 0.8 / 0.0
	A 1.7 / 0.0	B 2.2 / 25.0	F 0.0 / 0.0	A 1.5 / 12.5	F 0.0 / 0.0	F 0.1 / 0.0	C 2.0 / 12.5	C 0.3 / 0.0
		F 0.0 / 0.0	A 1.6 / 0.0	F 0.0 / 0.0	A 2.0 / 0.0	F 0.0 / 0.0	C 1.4 / 0.0	
			F 0.4 / 0.0	A 1.6 / 0.0	F 0.2 / 0.0	B 1.5 / 37.5	C 1.5 / 12.5	
			ASSEMBLY TYPE→	F 0.0 / 0.0	A 20.7 / 12.5	C 1.4 / 0.0	C 0.2 / 0.0	
					C 1.7 / 0.0	C 0.4 / 0.0		

←CALC. FAILURE PROB.
←OBSERVED FAILURE FREQ.

FIG. 7.17 COMPARISON OF CALCULATED ASSEMBLY FAILURE PROBABILITIES WITH OBSERVED FAILURE FREQUENCY DISTRIBUTION OVER A CORE OCTANT - CYCLE 1A. MAINE YANKEE

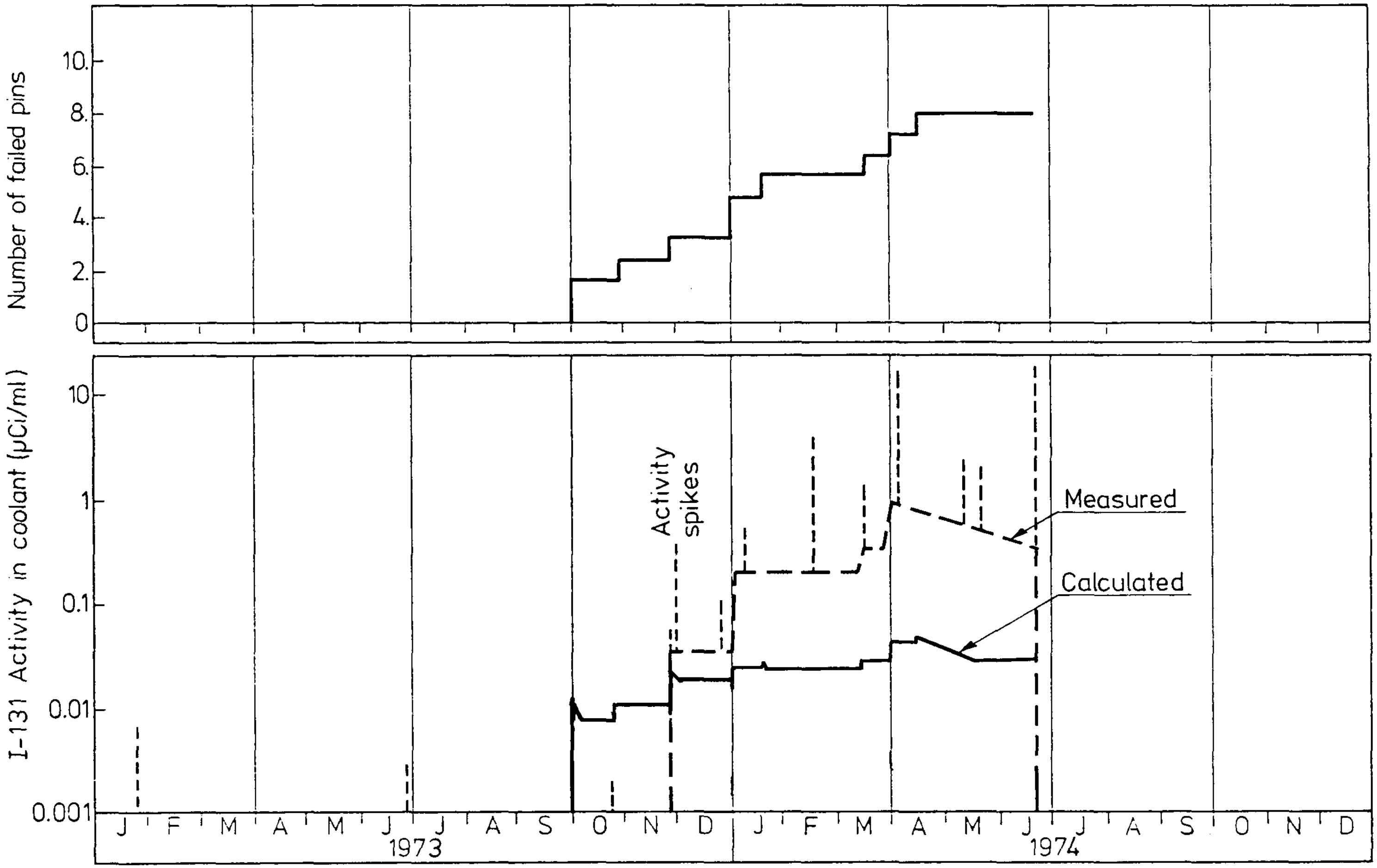


FIG. 7.18 GROSS CORE ACCUMULATED PREDICTED PIN FAILURE RATE COMPARED TO I-131 ACTIVITY IN COOLANT VERSUS TIME - CYCLE 1. MAINE YANKEE

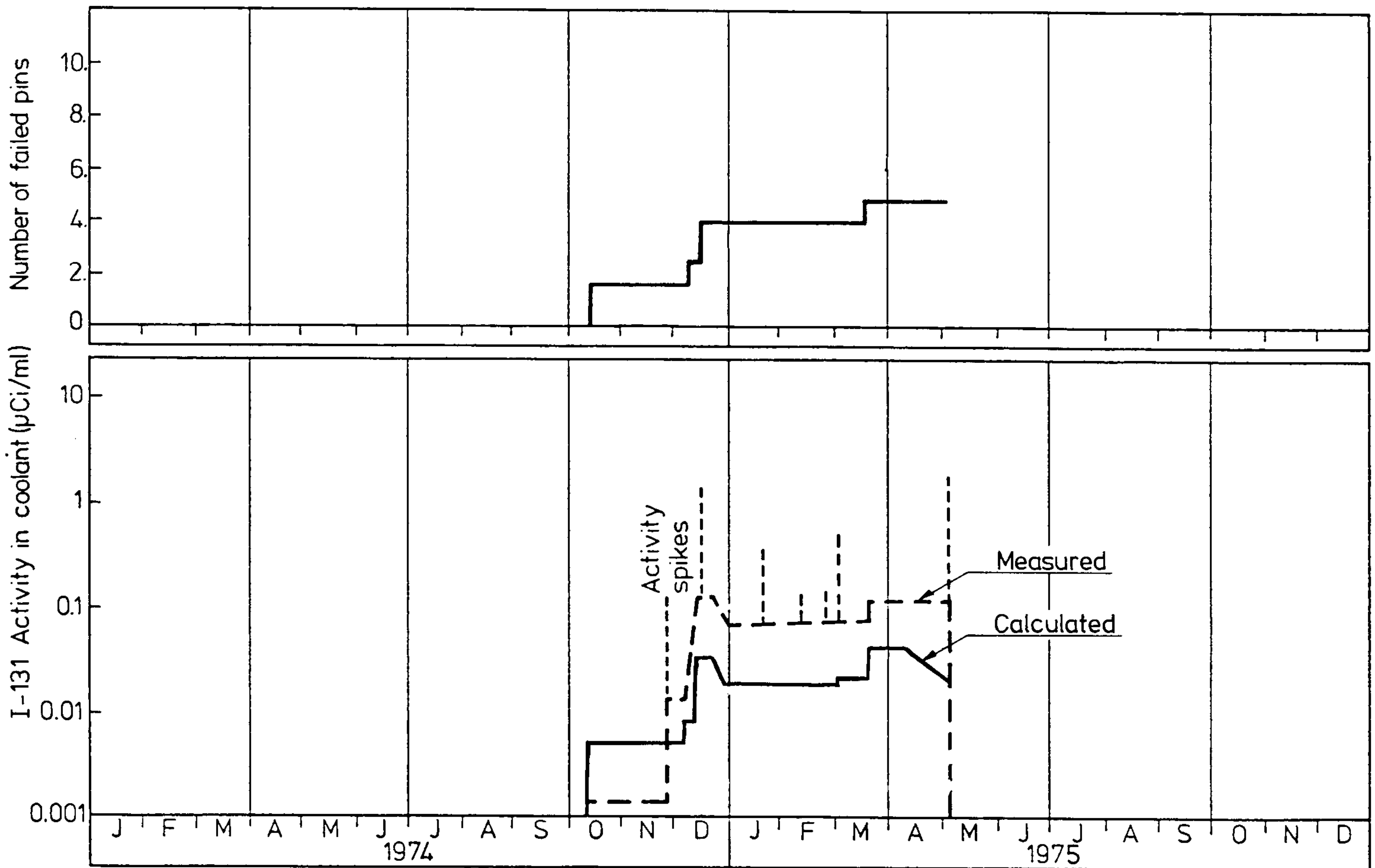


FIG. 7.19 GROSS CORE ACCUMULATED PREDICTED PIN FAILURE RATE COMPARED TO I-131 ACTIVITY IN COOLANT VERSUS TIME - CYCLE 1A. MAINE YANKEE

Section 8

CONCLUSIONS AND RECOMMENDATIONS

This study represents the first comprehensive application of POSHO in a nuclear fuel failure analysis based on detailed nodal power histories as a function of time. Comparison of the results of the analysis with the experience from Cycles 1 and 2 at Quad Cities Unit Two and Cycles 1 and 1A at Maine Yankee shows that the model can predict the approximate frequency of failures for the fuel in these reactors, except for the Type B fuel in Maine Yankee. Prediction of where the failures occur is possible when the failures are caused by relatively large power shocks (as occurred in Quad Cities Unit Two, Cycle 2 and during the rod withdrawal event of Cycle 1). If the failures are caused by a large number of moderate power shocks distributed throughout the core, actual locations of failures cannot be predicted.

The results from the use of the simple fission product release algorithm to predict activity levels are encouraging, but further development of the model is required to provide better accuracy and better information on the time of occurrence of failures.

Improvements in the POSHO model could be achieved by taking into account the change in failure probability resulting from power shock history (a cumulative damage factor), by adjustments based on more detailed information on the location of failed fuel pins and by a better understanding of the amount of preconditioning of the fuel that takes place during slow ramp power increases. Improvements in the fission product release algorithm could be achieved by including a burnup dependency for the release rate and by considering the activity release from fuel material present in the primary coolant system as a result of prior failures.

To make the failure model and the fission product release algorithm more effective, to understand the causes of power shocks that result in fuel failure and to locate the failed assemblies, it is recommended that 1) the data base be improved by acquisition of additional detailed nodal power histories and, more important, by acquisition of detailed information on the number of failed fuel pins and their locations;

2) the model be improved to take into account fuel pin shock history and the preconditioning effect during slow power ramp increases; 3) the fission product release algorithm be modified to take into account the deficiencies described above and the possible better correlations that could result from consideration of specific isotope activity levels and, 4) the acquisition of detailed data on primary coolant (PWR) and offgas (BWR) isotopic activity levels as a function of time.

Section 9

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3. S. Børresen, "A simplified, coarse mesh, three-dimensional diffusion scheme for calculating the gross power distribution in a boiling water reactor". Nucl. Sci. Eng., 44, 37-43 (1971).

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* HPR reports are available generally only to members of the Halden Reactor Project.

APPENDIX A

DESCRIPTION OF COMPUTER PROGRAMS

I THE FUEL MANAGEMENT SYSTEM - FMS

- System Description
- RECORD Programs
- PRESTO Program

II FUEL DUTY CYCLE ANALYSIS - FDCA

- Excerpts of FDCA Program Documentation

APPENDIX A

DESCRIPTION OF COMPUTER PROGRAMS

I. THE FUEL MANAGEMENT SYSTEM - FMS

The Fuel Management System (FMS) is a unique system developed for the specific purpose of meeting utility requirements for in-house fuel management analysis. The system consists of a complete computer program package for analysis, simulation, and optimization of the core performance of modern Light Water Reactors (LWRs).

Present practices for reactor core analysis rely heavily on trial and error procedures and the use of large, general purpose computer programs requiring many technical experts in various fields to perform a complete core analysis. FMS overcomes these requirements by providing:

- **Integrated computer models tailored to characteristics of LWR cores.**
- **Standard calculational procedures ensuring adequate treatment of important nuclear and thermodynamic effects at various levels of calculational sophistication.**
- **A user-oriented system emphasizing automation with a common data base to transfer data between programs and to access previous data.**
- **A systematic approach for optimal fuel management, including an overall optimization model.**

Using these features, FMS supports the main activities of nuclear fuel management as described in the previous section.

System Descriptions

FMS consists of four major computer programs – RECORD, FALC, FCS II, and PRESTO, which make use of the principle of superposition to separate reactor calculations into local and global calculations. The RECORD program provides detailed calculations for the local models. The FALC, FCS II, and PRESTO programs simulate reactor performance on three separate levels of sophistication. The DATA BANK provides automated communication between computer programs.

Each of these programs perform specific functions and is required to permit comprehensive fuel management analysis.

The RECORD program develops the reactor physics model of the fuel assemblies. It is a two-dimensional program which, as a function of burnup, develops a neutron energy spectrum for each fuel pin and coalesces multigroup neutron cross sections into few group cross sections for fuel assemblies. The RECORD program is specifically intended for investigation of fuel assembly design variations, including fuel enrichment, fuel pin radii, void fraction, burnable poisons, and plutonium island designs. Efficiency and high accuracy are provided by highly developed automation which requires a minimum of manual data transfer and computer time.

The FALC program performs investigations and economic comparisons of fuel cycle alternatives based on a reactor physics model of the fuel assemblies developed by RECORD. The FALC program develops, by linear programming techniques, optimum fuel loading schedules through as many as 12 refueling cycles for each set of alternatives and constraints.

FCS II performs one-dimensional reactor core simulation. It develops radial or axial power and reactivity distributions as a function of reactor burnup based on the two group reactor physics model of the fuel assemblies developed by RECORD. FCS II will also determine fuel cycle cost based on the isotopic concentrations of the fuel either calculated by the program or from input.

The PRESTO program performs three-dimensional reactor core simulations. It develops detailed power, void, and reactivity distributions as a function of reactor burnup based on a two group reactor physics model of the fuel assemblies developed by RECORD. It includes integrated thermal-hydraulics and neutronics models and auxiliary routines for generating reflector albedos using two-dimensional fine mesh diffusion theory. PRESTO also generates data for direct comparison with plots of traveling incore probes and other incore detectors.

Optimization

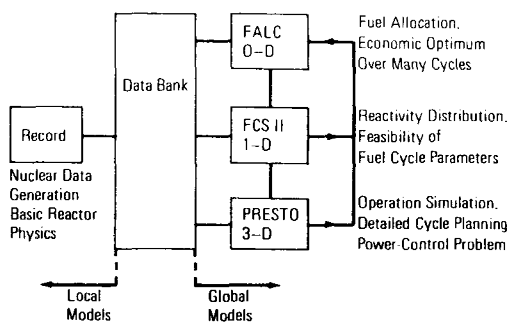
The FMS programs facilitate a systematic approach to optimum fuel utilization in the core. In this procedure, the FALC model is used to describe the reactor throughout several cycles of the reactor life. FALC is designed to give fast answers and a good relative comparison between various fuel cycle alternatives, based on the present worth weighted fuel cycle costs. Thus, many alternatives can be compared before committing a large nuclear analysis activity to any particular alternative.

Once the choice has been narrowed to a few selected alternatives, the field can be narrowed still further by surveying the performance feasibility of the various alternatives. FCS II is used to investigate the ability of the alternatives to meet cycle length and power distribution limits.

Complete reactor analysis in three dimensions can now be performed with PRESTO on the most attractive alternative for the operating cycle. The performance of this reactor analysis will attempt to demonstrate that the proposed alternative is feasible with respect to power distributions, control, safety limitations, and ability to sustain full operation of the reactor through the operating cycle.

This optimization procedure allows alternatives to be investigated with simple models in the beginning, and more sophisticated models as the number of alternatives is decreased. The final answer represents the optimum solution for a given reactor operating schedule within the constraints of the reactor operating limitations.

A schematic outline of FMS and the fuel cycle optimization procedure is shown below. Additional descriptions of the FMS programs and examples of their use are given in subsequent sections of this brochure.



FMS–SYSTEM

Schematic Outline of Fuel Cycle Optimization System and Procedure

RECORD Program

RECORD embodies the local models of the nuclear fuel assemblies which calculates the macroscopic nuclear constants as a function of fuel burnup, void fraction, and exposure weighted void fraction. The macroscopic constants necessary to describe the fuel assemblies are infinite multiplication factor, two and five group nuclear cross sections, and diffusion coefficients. RECORD also calculates the pin power and isotopic densities.

Fuel assemblies may be described as a two-dimensional square lattice of pins surrounded by a flow box with adjacent water gaps as shown in Figure 1. Each pin may have a different fuel or absorber composition. Control rods may be solid blade, rodded blade, or rod cluster control. The set of fuel and absorber materials available are sufficient to describe normal, light water reactor core materials. Gadolinium and soluble boron can be included as absorber materials. RECORD is specifically designed to handle fuel assembly heterogeneities, such as variable fuel enrichment, variable fuel pin radius, burnable poison pins, plutonium island designs, etc.

The method of solution is illustrated in Figure 2. In the epithermal range, the neutron spectrum is calculated for an average pin cell and assumed to be independent of spacial effects. Recalculation of the epithermal spectrum at specified burnup values is optional. The microscopic constants are determined for each isotope and are then combined with the appropriate isotopic densities at each time step to produce the macroscopic constants. The neutron spectrum is calculated in 35 groups with the B-1 approximation, applying extended Grueling-Goertzel slowing down. A modified Dancoff factor is applied to the resonance calculations to account for reduced shielding in edge and corner pins.

The thermal energy range extends up to 1.84 eV. Spectrum calculations are performed in this range by solving the transport equation with a development of the point energy approach. Neutron scattering is described by the Nelkin scattering model for hydrogen bound in water and the Brown-St. John model for oxygen. Energy dependent flux disadvantage factors, determined with a modified Amoyal-Benoist method, are used with the spectrum from the average pin cell to calculate the region-wise neutron spectra for each pin. This pin spectra is combined with multi-group material cross sections to produce the macroscopic constants for the thermal energy range. The thermal energy calculations are repeated for each burnup step.

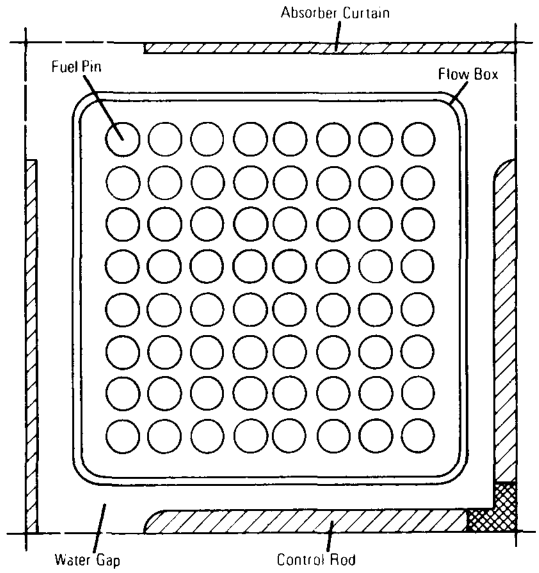


Figure 1. Typical BWR Assembly

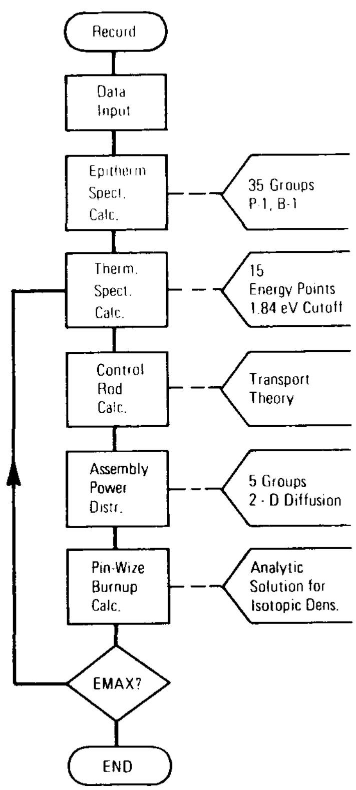


Figure 2. RECORD Method of Solution

Nuclear constants for control absorbers are determined using transport theory neutron reflection and transmission probabilities and include first order anisotropy and detailed geometrical representation.

Step-wise analytic solutions are used for the isotopic density variations in each fuel pin. Fission products are treated with eleven explicit fission products in four chains and four pseudo fission products. Void fraction, soluble boron concentration and the power density may be changed for each burnup step. Control rods, control rod followers, and absorber curtains may also be inserted or removed at each burnup step.

Flux and power distributions across the fuel assembly are calculated by a 5-group finite difference diffusion routine employing a modified Wielandt technique to give fast convergence. These neutron fluxes are used to obtain the assembly averaged macroscopic nuclear constants. For fuel pins containing gadolinium as a burnable poison, the effective thermal cross sections are generated off-line by a special burnup version of THERMOS. RECORD uses data in the form of polynomial functions of void and cell surface flux for both thermal groups. In the epithermal region the pins are treated as regular pins.

Options

Calculations may be performed either for a single pin or for a fuel assembly.

Restrictions

Maximum pin array:	9x9 (extendable)
Maximum thermal microgroups:	15 (extendable)

Running Time

The computer time used by the RECORD program will vary from case to case. A typical 7 x 7 BWR assembly with 22 x 22 mesh points requires about 35 seconds for the initial calculation and 23 seconds per burnup step when run on the CDC CYBER 74 computer (approximately equivalent to a CDC 6600).

Programming Language: FORTRAN

Memory Requirements

The estimated central memory required to run the RECORD program is:

	CDC Words	IBM Bytes
Central Memory	41.000	190.000

PRESTO Program

PRESTO is a light water reactor simulator, designed to provide three-dimensional life time histories of power, reactivity, and flow distributions. PRESTO provides integrated thermal-hydraulics and neutronics models for this simulation.

Three-dimensional core burnup histories can be calculated for any sequence of reactor operating data, including changes in core power and control rod insertion patterns. Refueling operation, such as loading of fresh fuel, reloading and discharge of exposed fuel, and fuel shuffling may be included in the simulation. PRESTO may also be used to calculate a generalized Haling power distribution. The present version calculates the steady state steam void distribution in the core for the BWR power range.

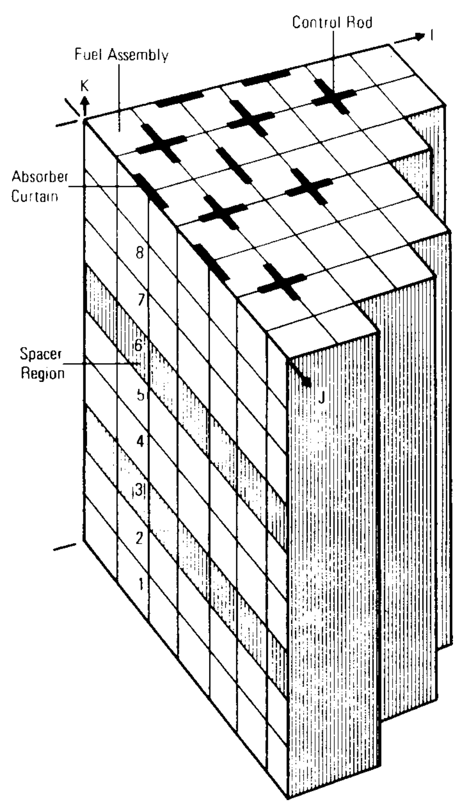


Figure 6. Typical BWR Geometry

A typical BWR geometrical configuration for PRESTO is shown in Figure 6. Fuel assemblies and control rods or control rod banks are individually represented. Burnable poison curtains positioned between fuel assemblies and the effect of axially located spacer grids may be included.

The method of solution is illustrated in Figure 7. A specially developed two-group diffusion scheme is used to calculate flux and power distributions in three dimensions. A finite difference formulation is used for the fast flux equation; a smoothed asymptotic distribution is assumed for the thermal flux.

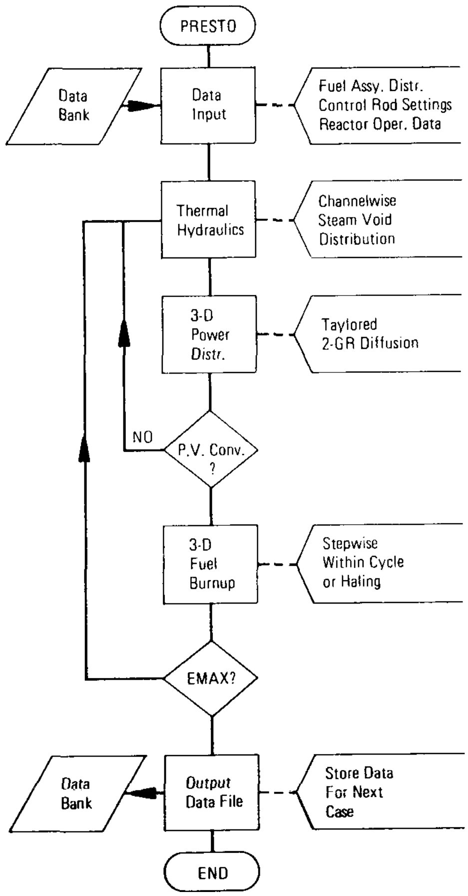


Figure 7. PRESTO Method of Solution

The two-group macroscopic cross-sections are provided as fitted polynominals which can be obtained from RECORD. Control rods, absorber curtains, and spacer grids are taken into account by additional polynomial fits. Equilibrium xenon concentration and Doppler effects are obtained as a function of local power density.

Feedback mechanisms, such as effect of local equilibrium xenon and Doppler effect, are included in the power distribution calculation. A Xe-dynamics routine is available for the analysis of power distributions following changes in core power or control rod positions. Core reactivity may be kept constant with flow control. This facility makes PRESTO well suited for operations guidance. If the Xe-dynamics option is selected, the time dependent Xe and I equations are solved useing either a constant flux or a linear flux extrapolation within each time step.

A total of 13 different core symmetry options are available. These include: Full core, half core, one quarter core, and one eighth core representations with either mirror or rotational symmetry and full or half node boundaries.

A special, fast thermal-hydraulics model was developed for PRESTO, which uses distributed power, feedwater ènthalpy, pump head, and zone throttling coefficients. The hydraulics module may also be bypassed and input void distributions used.

One special feature of the PRESTO program allows the use of a built-in two-dimensional diffusion routine to calculate albedos for the core boundaries. Another special feature calculates results for direct comparison with measurements from incore probes. Both calculated and measured results can be plotted on the same graph by PRESTO.

Options

Reactor performance may be simulated in one, two, or three dimensions.

Restrictions

Maximum number of nodes:	14,400
Maximum number of channels:	600
Maximum number of control rods:	150
Maximum number of channels in x or y direction:	30
Maximum number of axial nodes:	24
Maximum number of throttling zones:	5
Maximum number of fuel types:	6

Running Time

The computer time used by the PRESTO program will vary widely from case to case. A typical 800 MW BWR quarter core calculation requires about 12 seconds per power-void iteration, and about 60 seconds per power-void iteration is required for the full core representation. These computer times are for a CDC CYBER 74 computer with 128,000 words of storage (approximately equivalent to a CDC 6600).

Programming Language: FORTRAN

Memory Requirements

The memory required to run the PRESTO program varies widely from case to case. Two estimates of these requirements are:

	CDC Words	IBM Bytes
Quarter core version	52,000	250,000
Full core version	90,000	460,000

II. FUEL DUTY CYCLE ANALYSIS - FDCA

Introduction

ScP has developed an empirical model, POSHO, for prediction of the probability for fuel failure in a fuel rod resulting from pellet clad interaction (PCI) during power ramps of the rod. This model requires as input detailed information about the "power shocks" created in the reactor core during operation. Such information may be deduced from information on local pin powers within the assembly and nodal power distribution as a function of time. In an operating plant, the latter may be obtained from the process computer, but it may also be obtained using the reactor simulation capability of FMS. The program package FDCA is an extension of FMS, incorporating POSHO into the overall model, thus being able to predict fuel failures in the core as a function of time. The FDCA model has been developed as standalone programs for eventual use in another environment than FMS, for example on the process computer of an operating plant.

Th. normal mode of application of the FDCA modules within the FMS environment is described below.

FDCA program package

The following programs are included:

PINGR - calculates average power density (w/cm) for a group of fuel pins within a fuel assembly based on individual pin power distributions as calculated by the FMS program RECORD. The pin-group power distributions are functions of exposure, void, exposure-weighted void and control fraction. These functions are represented by polynomial fits using the FMS-module POLGEN.

FDCA1 - calculates "power shocks" (ΔQ) and "zero gap power densities" (P') for each pin-group of each node in the core based on information on the reactor state variables (stable conditions) before and after each "power shock event."

The stable operation periods are normally represented as time steps or burnup steps in reactor simulation (PRESTO). A simulated event to be treated by FDCA1, is thus defined by the transition from step j-1 to step j in a series of PRESTO calculations. Calculational results of FDCA1, for a sequence of events, are stored on a magnetic tape called the Q-file.

FDCA2 - calculates PCI fuel failure probabilities for a given sequence of (ΔQ, Q) distributions as obtained from a Q-file produced by FDCA1. Such failure probabilities are calculated for each fuel pin-group of each node. Failure probabilities for pins, assemblies and regions and for the entire core as well as expected number of failed pins, failed assemblies and expected no. of cracks per pin, per assembly etc. are then calculated. These quantities are also accumulated in time over a selection of the time steps contained on the Q-file.

A schematic overview of the FMS/FDCA program system is shown in Fig. 1.

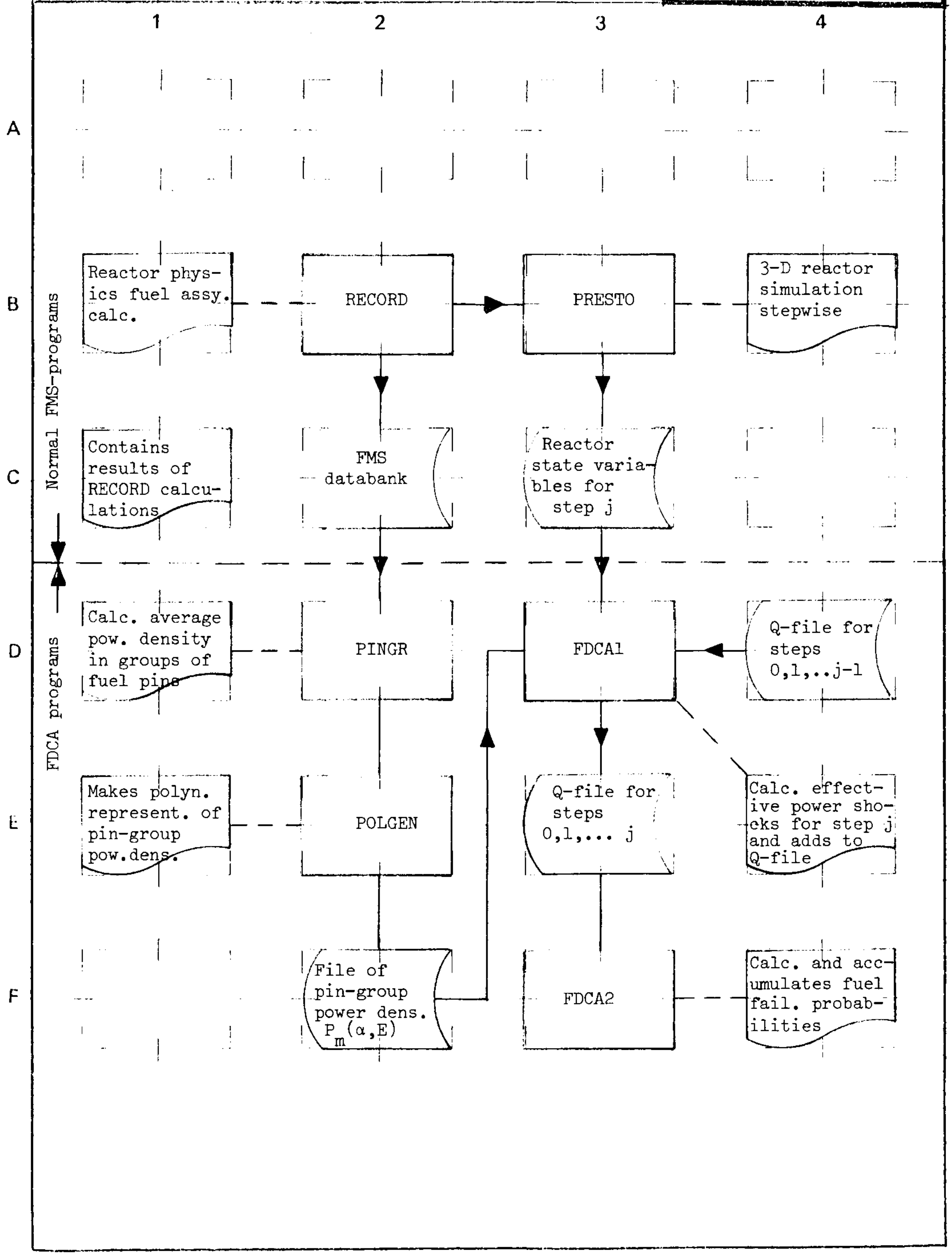
F L O W C H A R T

Routine: DATAFLOW

Part:

Author: T.O. Sauar/S. Børresen

FMS/1/ SYSTEM	/B-02
Institutt for Atomenergi	
Page : 3	
Date : 22.12.1975	



APPENDIX B

DATA SETS FOR

QUAD CITIES 2 SIMULATION

QUAD CITIES - UNIT 2

ROD PATTERN:

1 8	2	3 13	4	5 7	6	7 10	8
9	10 24	11	12 24	13	14 21	15	16
17 13	18	19 8	20	21 11	22	23 9	24
25	26 24	27	28 20	29	30 24	31	
32 7	33	34 11	35	36 6	37	38	
39	40 21	41	42 24	43	44		
45 10	46	47 9	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC2730103
OPR. CYCLE	1
DATE	01/03/73
ACC. BURNUP	1474 MWD/TU
TH. POWER	2447.8 MW
FLOW RATE	$1.234 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.900 \cdot 10^4$ J/KG
TIP SET 1	BOC1 (FIG 6.17)

ROD PATTERN:

1	2 6	3	4 12	5	6 6	7	8
9 24	10	11 22	12	13 20	14	15 20	16
17	18 14	19	20 6	21	22 15	23	24
25 24	26	27 20	28	29 23	30	31 9	
32	33 6	34	35 10	36	37 5	38	
39 20	40	41 22	42	43 20	44		
45	46 10	47	48	49			
50 15	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC2730123
OPR. CYCLE	1
DATE	01/23/73
ACC. BURNUP	1719 MWD/TU
TH. POWER	2445.8 MW
FLOW RATE	$1.230 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.868 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2 8	3	4 14	5	6 8	7	8
9 24	10	11 24	12	13 24	14	15 24	16
17	18 15	19	20 8	21	22 15	23	24
25 24	26	27 24	28	29 24	30	31 11	
32	33 8	34	35 12	36	37 8	38	
39 24	40	41 24	42	43 24	44		
45	46 10	47	48	49			
50 24	51	52 6					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2730122
OPR. CYCLE	1
DATE	01/22/73
ACC. BURNUP	1719 MWD/TU
TH. POWER	2243.4 MW
FLOW RATE	$1.235 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.698 \cdot 10^4$ J/KG

ROD PATTERN:

1	2 16	3	4 2	5	6 12	7	8
9 24	10	11 24	12	13 22	14	15 20	16
17	18 5	19	20 12	21	22 7	23	24
25 24	26	27 22	28	29 22	30	31	
32	33 16	34	35 2	36	37 10	38	
39 22	40	41 22	42	43 18	44		
45	46 6	47	48	49			
50 15	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2730320
OPR. CYCLE	1
DATE	03/20/73
ACC. BURNUP	2536 MWD/TU
TH. POWER	2378.8 MW
FLOW RATE	$1.225 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.742 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2 16	3	4	5	6 12	7	8
9 24	10	11 24	12	13 22	14	15 20	16
17	18 5	19	20 12	21	22 5	23	24
25 24	26	27 22	28	29 22	30	31	
32	33 15	34	35 3	36	37 10	38	
39 22	40	41 22	42	43 18	44		
45	46 4	47	48	49			
50 18	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2730410
OPR. CYCLE	1
DATE	04/10/73
ACC. BURNUP	2874 MWD/TU
TH. POWER	2378.8 MW
FLOW RATE	$1.225 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.742 \cdot 10^4$ J/KG

ROD PATTERN:

1 24	2	3 24	4	5 22	6	7 22	8
9	10 8	11	12 16	13	14 5	15	16
17 24	18	19 22	20	21 22	22	23 24	24
25	26 16	27	28 4	29	30 10	31	
32 22	33	34 22	35	36 20	37	38	
39	40 5	41	42 10	43	44		
45 22	46	47 24	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 27305071
OPR. CYCLE	1
DATE	05/07/73
ACC. BURNUP	2989 MWD/TU
TH. POWER	2450.0 MW
FLOW RATE	$1.230 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.918 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1 24	2	3 24	4	5 22	6	7 22	8
9	10 8	11	12 16	13	14 5	15	16
17 24	18	19 22	20	21 22	22	23 20	24
25	26 16	27	28 3	29	30 10	31	
32 22	33	34 22	35	36 20	37	38	
39	40 5	41	42 10	43	44		
45 22	46	47 20	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 27305072
OPR. CYCLE	1
DATE	05/07/73
ACC. BURNUP	2989 MWD/TU
TH. POWER	2481.0 MW
FLOW RATE	$1.203 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.918 \cdot 10^4$ J/KG

ROD PATTERN:

1 22	2	3 22	4	5 22	6	7 22	8
9	10 8	11	12 25	13	14 5	15	16
17 22	18	19 21	20	21 22	22	23 20	24
25	26 16	27	28 4	29	30 10	31	
32 22	33	34 22	35	36 19	37	38	
39	40 5	41	42 10	43	44		
45 22	46	47 22	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2730510
OPR. CYCLE	1
DATE	05/10/73
ACC. BURNUP	3037 MWD/TU
TH. POWER	2378.3 MW
FLOW RATE	$1.225 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.765 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1 24	2	3 22	4	5 22	6	7 21	8
9	10 9	11	12 14	13	14 7	15	16
17 22	18	19 22	20	21 21	22	23 19	24
25	26 14	27	28 5	29	30 13	31	
32 21	33	34 21	35	36 20	37	38	
39	40 7	41	42 13	43	44		
45 21	46	47 19	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2730625
OPR. CYCLE	1
DATE	06/25/73
ACC. BURNUP	3608 MWD/TU
TH. POWER	2358.3 MW
FLOW RATE	$1.186 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.914 \cdot 10^4$ J/KG

ROD PATTERN:

1 24	2	3 22	4	5 22	6	7 21	8
9	10 9	11	12 14	13	14 6	15	16
17 22	18	19 22	20	21 21	22	23 19	24
25	26 14	27	28 4	29	30 13	31	
32 21	33	34 21	35	36 20	37	38	
39	40 6	41	42 13	43	44		
45 21	46	47 19	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2730726
OPR. CYCLE	1
DATE	07/26/73
ACC. BURNUP	4110 MWD/TU
TH. POWER	2358.3 MW
FLOW RATE	$1.186 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.914 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

←Control rod no. - PRESTO
←Rod insertion - nodes

1 24	2	3 22	4	5 21	6	7 20	8
9	10 9	11	12 14	13	14 7	15	16
17 22	18	19 22	20	21 21	22	23 19	24
25	26 14	27	28 5	29	30 13	31	
32 21	33	34 21	35	36 20	37	38	
39	40 7	41	42 14	43	44		
45 20	46	47 20	48	49			
50	51	52					

DATASET NO.	QC 2730817
OPR. CYCLE	1
DATE	08/17/73
ACC. BURNUP	4283 MWD/TU
TH. POWER	2344.7 MW
FLOW RATE	$1.218 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.756 \cdot 10^4$ J/KG

ROD PATTERN:

←Control rod no. - PRESTO
←Rod insertion - nodes

1	2 24	3	4 21	5	6 20	7	8
9 5	10	11 16	12	13 6	14	15 15	16
17	18 21	19	20 21	21	22 20	23	24
25 16	26	27 7	28	29 16	30	31	
32	33 21	34	35 20	36	37 20	38	
39 5	40	41 16	42	43 4	44		
45	46 20	47	48 19	49			
50	51	52					

DATASET NO.	QC 2730914
OPR. CYCLE	1
DATE	09/14/73
ACC. BURNUP	4797 MWD/TU
TH. POWER	2248.6 MW
FLOW RATE	$1.142 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.884 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2 24	3	4 24	5	6 21	7	8
9 8	10	11 13	12	13 8	14	15 12	16
17	18 22	19	20 22	21	22 21	23	24
25 13	26	27 6	28	29 12	30	31	
32	33 21	34	35 22	36	37 21	38	
39 7	40	41 12	42	43 3	44		
45	46 22	47	48 21	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2730920
OPR. CYCLE	1
DATE	09/20/73
ACC. BURNUP	4797 MWD/TU
TH. POWER	2245.0 MW
FLOW RATE	$1.170 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.730 \cdot 10^4$ J/KG

ROD PATTERN:

1	2 24	3	4 24	5	6 21	7	8
9 8	10	11 13	12	13 8	14	15 12	16
17	18 22	19	20 22	21	22 21	23	24
25 13	26	27 6	28	29 12	30	31	
32	33 21	34	35 22	36	37 21	38	
39 7	40	41 12	42	43 3	44		
45	46 22	47	48 21	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2730922
OPR. CYCLE	1
DATE	09/22/73
ACC. BURNUP	4895 MWD/TU
TH. POWER	2245.0 MW
FLOW RATE	$1.170 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.730 \cdot 10^4$ J/KG
	Xe-trans

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2 22	3	4 20	5	6 20	7	8
9 8	10	11 14	12	13 7	14	15 12	16
17	18 24	19	20 21	21	22 20	23	24
25 13	26	27 6	28	29 12	30	31	
32	33 21	34	35 20	36	37 19	38	
39 5	40	41 12	42	43 3	44		
45	46 20	47	48 19	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2731221
OPR. CYCLE	1
DATE	12/21/73
ACC. BURNUP	6081 MWD/TU
TH. POWER	2266.9 MW
FLOW RATE	$1.224 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.551 \cdot 10^4$ J/KG

ROD PATTERN:

1	2 24	3	4 22	5	6 24	7	8
9 13	10	11 5	12	13 11	14	15 5	16
17	18 24	19	20 21	21	22 21	23	24
25 3	26	27 11	28	29 2	30	31	
32	33 24	34	35 24	36	37 18	38	
39 9	40	41 3	42	43 9	44		
45	46 19	47	48 15	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2731227
OPR. CYCLE	1
DATE	12/27/73
ACC. BURNUP	6162 MWD/TU
TH. POWER	2255.3 MW
FLOW RATE	$1.233 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.488 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2 24	3	4 21	5	6 24	7	8
9 13	10	11 5	12	13 11	14	15 5	16
17	18 24	19	20 21	21	22 21	23	24
25 3	26	27 11	28	29 2	30	31	
32	33 24	34	35 21	36	37 18	38	
39 9	40	41 3	42	43 9	44		
45	46 19	47	48 15	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740102
OPR. CYCLE	1
DATE	01/02/74
ACC. BURNUP	6321 MWD/TU
TH. POWER	2349.4 MW
FLOW RATE	$1.199 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.840 \cdot 10^4$ J/KG

ROD PATTERN:

1	2 21	3	4 21	5	6 21	7	8
9 13	10	11 5	12	13 11	14	15 5	16
17	18 22	19	20 21	21	22 20	23	24
25 3	26	27 11	28	29 3	30	31	
32	33 21	34	35 21	36	37 18	38	
39 9	40	41 4	42	43 9	44		
45	46 19	47	48 15	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740208
OPR. CYCLE	1
DATE	02/08/74
ACC. BURNUP	6966 MWD/TU
TH. POWER	2217.6 MW
FLOW RATE	$1.232 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.451 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2 / 24	3	4 / 24	5	6 / 24	7	8
9 / 10	10	11 / 3	12	13 / 9	14	15 / 3	16
17	18 / 24	19	20 / 24	21	22 / 24	23	24
25	26	27 / 7	28	29	30	31	
32	33 / 24	34	35 / 24	36	37 / 14	38	
39 / 6	40	41 / 2	42	43 / 5	44		
45	46 / 17	47	48 / 10	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 27402121
OPR. CYCLE	1
DATE	02/12/74
ACC. BURNUP	7000 MWD/TU
TH. POWER	1880.0 MW
FLOW RATE	$8.870 \cdot 10^3$ KG/SEC
SUBCOOLING	$5.721 \cdot 10^4$ J/KG

ROD PATTERN:

1	2 / 24	3	4 / 24	5	6 / 24	7	8
9 / 10	10	11 / 3	12	13 / 9	14	15 / 3	16
17	18 / 24	19	20 / 18	21	22 / 24	23	24
25	26	27 / 7	28	29	30	31	
32	33 / 24	34	35 / 24	36	37 / 14	38	
39 / 6	40	41 / 2	42	43 / 5	44		
45	46 / 17	47	48 / 10	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 27402122
OPR. CYCLE	1
DATE	02/12/74
ACC. BURNUP	7000 MWD/TU
TH. POWER	1992.9 MW
FLOW RATE	$8.928 \cdot 10^3$ KG/SEC
SUBCOOLING	$5.721 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2: 24	3	4: 24	5	6: 24	7	8
9: 10	10	11: 3	12	13: 9	14	15: 3	16
17	18: 24	19	20: 19	21	22: 24	23	24
25	26	27: 7	28	29	30	31	
32	33: 24	34	35: 24	36	37: 14	38	
39: 6	40	41: 2	42	43: 5	44		
45	46: 17	47	48: 10	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 740214
OPR. CYCLE	1
DATE	02/14/74
ACC. BURNUP	7029 MWD/TU
TH. POWER	2252.3 MW
FLOW RATE	$1.202\cdot10^4$ KG/SEC
SUBCOOLING	$4.644\cdot10^4$ J/KG

ROD PATTERN:

1	2: 21	3	4: 21	5	6: 21	7	8
9: 10	10	11: 3	12	13: 9	14	15: 3	16
17	18: 24	19	20: 19	21	22: 20	23	24
25	26	27: 7	28	29	30	31	
32	33: 22	34	35: 21	36	37: 14	38	
39: 6	40	41: 2	42	43: 5	44		
45	46: 17	47	48: 10	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 740320
OPR. CYCLE	1
DATE	03/20/74
ACC. BURNUP	7480 MWD/TU
TH. POWER	2319.9 MW
FLOW RATE	$1.233\cdot10^4$ KG/SEC
SUBCOOLING	$4.677\cdot10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1 8	2	3 3	4	5 9	6	7 2	8
9	10 24	11	12 20	13	14 24	15	16
17 3	18	19 10	20	21 2	22	23 7	24
25	26 21	27	28 20	29	30 17	31	
32 9	33	34 2	35	36 6	37	38	
39	40 24	41	42 17	43	44		
45 2	46	47 7	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740405
OPR. CYCLE	1
DATE	04/05/74
ACC. BURNUP	7601 MWD/TU
TH. POWER	2374.1 MW
FLOW RATE	$1.238 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.754 \cdot 10^4$ J/KG

ROD PATTERN:

1 8	2	3 3	4	5 9	6	7 4	8
9	10 24	11	12 20	13	14 24	15	16
17 3	18	19 10	20	21 2	22	23 7	24
25	26 21	27	28 20	29	30 17	31	
32 9	33	34 2	35	36 6	37	38	
39	40 24	41	42 17	43	44		
45 4	46	47 7	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740415
OPR. CYCLE	1
DATE	04/15/74
ACC. BURNUP	7671 MWD/TU
TH. POWER	2374.1 MW
FLOW RATE	$1.238 \cdot 10^4$ KG/SEC
SUBCOOLING	J/KG
	Xe-trans

QUAD CITIES - UNIT 2

ROD PATTERN:

1 8	2	3 3	4	5 9	6	7 2	8
9	10 21	11	12 20	13	14 21	15	16
17 3	18	19 10	20	21 2	22	23 7	24
25	26 20	27	28 20	29	30 17	31	
32 9	33	34 2	35	36 6	37	38	
39	40 21	41	42 17	43	44		
45 2	46	47 7	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740501
OPR. CYCLE	1
DATE	05/01/74
ACC. BURNUP	7937 MWD/TU
TH. POWER	2467.5 MW
FLOW RATE	$1.222 \cdot 10^4$ KG/SEC
SUBCOOLING	$5.021 \cdot 10^4$ J/KG

ROD PATTERN:

1 5	2	3 2	4	5 4	6	7	8
9	10 24	11	12 19	13	14 24	15	16
17 2	18	19 5	20	21	22	23 4	24
25	26 19	27	28 22	29	30 12	31	
32 4	33	34	35	36 3	37	38	
39	40 24	41	42 12	43	44		
45	46	47 4	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740523
OPR. CYCLE	1
DATE	05/23/74
ACC. BURNUP	8294 MWD/TU
TH. POWER	2447.7 MW
FLOW RATE	$1.229 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.972 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1 6	2	3 2	4	5 4	6	7 2	8
9	10 24	11	12 20	13	14 22	15	16
17 2	18	19 5	20	21 2	22	23 4	24
25	26 20	27	28 21	29	30 13	31	
32 4	33	34 2	35	36 3	37	38	
39	40 22	41	42 13	43	44		
45 2	46	47 4	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740607
OPR. CYCLE	1
DATE	06/07/74
ACC. BURNUP	8456 MWD/TU
TH. POWER	2228.3 MW
FLOW RATE	$1.175 \cdot 10^{3}$ KG/SEC
SUBCOOLING	$4.958 \cdot 10^{3}$ J/KG

ROD PATTERN:

1	2	3	4 4	5	6	7	8
9 21	10	11 20	12	13 18	14	15 20	16
17	18 8	19	20	21	22 4	23	24
25 20	26	27 21	28	29 20	30	31 14	
32	33	34	35 5	36	37	38	
39 20	40	41 17	42	43 15	44		
45	46 5	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740709
OPR. CYCLE	1
DATE	07/09/74
ACC. BURNUP	8628 MWD/TU
TH. POWER	2251.4 MW
FLOW RATE	$1.240 \cdot 10^{4}$ KG/SEC
SUBCOOLING	$4.691 \cdot 10^{4}$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3	4	5	6	7	8
9 21	10	11 18	12	13 11	14	15 20	16
17	18 4	19	20	21	22 3	23	24
25 20	26	27 21	28	29 20	30	31	
32	33	34	35 2	36	37	38	
39 20	40	41 12	42	43 9	44		
45	46 2	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740809
OPR. CYCLE	1
DATE	08/09/74
ACC. BURNUP	9173 MWD/TU
TH. POWER	2335.7 MW
FLOW RATE	$1.225 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.888 \cdot 10^4$ J/KG

ROD PATTERN:

1	2	3	4	5	6	7	8
9 24	10	11 13	12	13 6	14	15 24	16
17	18 2	19	20	21	22	23	24
25 24	26	27 24	28	29 24	30	31	
32	33	34	35	36	37	38	
39 24	40	41 8	42	43 3	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740812
OPR. CYCLE	1
DATE	08/12/74
ACC. BURNUP	9203 MWD/TU
TH. POWER	2078.0 MW
FLOW RATE	$9.465 \cdot 10^3$ KG/SEC
SUBCOOLING	$5.765 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3	4	5	6	7	8
9 24	10	11 13	12	13 6	14	15 18	16
17	18 2	19	20	21	22	23	24
25 24	26	27 24	28	29 24	30	31	
32	33	34	35	36	37	38	
39 24	40	41 8	42	43 3	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2740813
OPR. CYCLE	1
DATE	08/13/74
ACC. BURNUP	9203 MWD/TU
TH. POWER	2078.0 MW
FLOW RATE	$9.465 \cdot 10^{3}$ KG/SEC
SUBCOOLING	$5.765 \cdot 10^{4}$ J/KG

ROD PATTERN:

1	2	3	4	5	6	7	8
9 24	10	11 24	12	13 5	14	15 12	16
17	18	19	20	21	22	23	24
25 11	26	27 24	28	29 24	30	31	
32	33	34	35	36	37	38	
39 15	40	41 5	42	43 4	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2741013
OPR. CYCLE	1
DATE	10/13/74
ACC. BURNUP	9778 MWD/TU
TH. POWER	2153.0 MW
FLOW RATE	$1.030 \cdot 10^{4}$ KG/SEC
SUBCOOLING	$5.349 \cdot 10^{4}$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3	4	5	6	7	8
9 24	10	11 15	12	13 5	14	15 12	16
17	18 2	19	20	21	22	23	24
25 11	26	27 22	28	29 24	30	31	
32	33	34	35	36	37	38	
39 15	40	41 5	42	43 4	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2741114
OPR. CYCLE	1
DATE	11/14/74
ACC. BURNUP	10258 MWD/TU
TH. POWER	2245.3 MW
FLOW RATE	$1.156 \cdot 10^4$ KG/SEC
SUBCOOLING	$5.107 \cdot 10^4$ J/KG

ROD PATTERN:

1 24	2	3 12	4	5 20	6	7 24	8
9	10	11	12	13	14 2	15	16
17 12	18	19 16	20	21 19	22	23	24
25	26	27	28	29	30	31	
32 20	33	34 19	35	36 3	37	38	
39	40 2	41	42	43	44		
45 24	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2741122
OPR. CYCLE	1
DATE	11/22/74
ACC. BURNUP	10314 MWD/TU
TH. POWER	1899.8 MW
FLOW RATE	$9.946 \cdot 10^3$ KG/SEC
SUBCOOLING	$5.133 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1 5	2	3 10	4	5 3	6	7 24	8
9	10	11	12	13	14	15	16
17 10	18	19 4	20	21 24	22	23	24
25	26	27	28	29	30	31	
32 3	33	34 24	35	36 2	37	38	
39	40	41	42	43	44		
45 24	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2741125
OPR. CYCLE	1
DATE	11/25/74
ACC. BURNUP	10346 MWD/TU
TH. POWER	1808.2 MW
FLOW RATE	$7.932 \cdot 10^3$ KG/SEC
SUBCOOLING	$6.298 \cdot 10^4$ J/KG

ROD PATTERN:

1 5	2	3 12	4	5 3	6	7 24	8
9	10	11	12	13	14	15	16
17 12	18	19 4	20	21 20	22	23	24
25	26	27	28	29	30	31	
32 3	33	34 21	35	36 2	37	38	
39	40	41	42	43	44		
45 24	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2741220
OPR. CYCLE	1
DATE	12/20/74
ACC. BURNUP	10697 MWD/TU
TH. POWER	1697.6 MW
FLOW RATE	$6.804 \cdot 10^3$ KG/SEC
SUBCOOLING	$6.870 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3 1	4	5	6	7 2	8
9	10 24	11	12 15	13	14 24	15	16
17 1	18	19	20	21	22	23	24
25	26 14	27	28 24	29	30 7	31	
32	33	34	35	36	37	38	
39	40 24	41	42 7	43	44		
45 2	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2750509
OPR. CYCLE	2
DATE	05/09/75
ACC. BURNUP	121 MWD/TU
TH. POWER	2031 MW
FLOW RATE	$1.048 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.551 \cdot 10^4$ J/KG

ROD PATTERN:

1	2	3	4	5	6	7	8
9	10 6	11	12 24	13	14 23	15	16
17	18	19	20	21	22	23	24
25	26 24	27	28 14	29	30 2	31	
32	33	34	35	36	37	38	
39	40 24	41	42 2	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC2750517
OPR. CYCLE	2
DATE	05/17/75
ACC. BURNUP	239 MWD/TU
TH. POWER	2358 MW
FLOW RATE	$1.200 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.323 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3 6	4	5	6	7 3	8
9	10 24	11	12 24	13	14 24	15	16
17 6	18	19	20	21	22	23	24
25	26 24	27	28 24	29	30 12	31	
32	33	34	35	36	37	38	
39	40 24	41	42 12	43	44		
45 3	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 27505211
OPR. CYCLE	2
DATE	05/21/75 19^{50}hrs.
ACC. BURNUP	264 MWD/TU
TH. POWER	1406 MW
FLOW RATE	6800.0 KG/SEC
SUBCOOLING	$5.58 \cdot 10^4$ J/KG

ROD PATTERN:

1	2	3	4	5	6	7	8
9	10 24	11	12 24	13	14 24	15	16
17	18	19	20	21	22	23	24
25	26 24	27	28 24	29	30 6	31	
32	33	34	35	36	37	38	
39	40 24	41	42 6	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 27505212
OPR. CYCLE	2
DATE	05/21/75 20^{45}hrs
ACC. BURNUP	264 MWD/TU
TH. POWER	1580 MW
FLOW RATE	7300 KG/SEC
SUBCOOLING	$5.58 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3	4	5	6	7	8
9	10 21	11	12 24	13	14 24	15	16
17	18	19	20	21	22	23	24
25	26 24	27	28 24	29	30 2	31	
32	33	34	35	36	37	38	
39	40 24	41	42 2	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 27505231
OPR. CYCLE	2
DATE	05/21/75 21^{05}hrs
ACC. BURNUP	264 MWD/TU
TH. POWER	1707 MW
FLOW RATE	8518 KG/SEC
SUBCOOLING	$5.349 \cdot 10^4$ J/KG

ROD PATTERN:

1	2	3	4	5	6	7	8
9	10 21	11	12 24	13	14 24	15	16
17	18	19	20	21	22	23	24
25	26 24	27	28 24	29	30 6*	31	
32	33	34	35	36	37	38	
39	40 24	41	42 2	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

* Insertion = 2 for quadrant 2.

DATASET NO.	QC 27505214
OPR. CYCLE	2
DATE	05/21/75 23^{30}hrs
ACC. BURNUP	264 MWD/TU
TH. POWER	2125 MW
FLOW RATE	$1.071 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.884 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3	4	5	6	7	8
9	10 21	11	12 24	13	14 24	15	16
17	18	19	20	21	22	23	24
25	26 24	27	28 24	29	30 4	31	
32	33	34	35	36	37	38	
39	40 24	41	42 2	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2750522
OPR. CYCLE	2
DATE	05/22/75 00^{20}hrs
ACC. BURNUP	264 MWD/TU
TH. POWER	2125 MW
FLOW RATE	$1.184 \cdot 10^4$ KG/SEC
SUBCOOLING	$4.884 \cdot 10^4$ J/KG

ROD PATTERN:

1	2	3	4	5	6	7	8
9	10 7	11	12 24	13	14 24	15	16
17	18	19	20	21	22	23	24
25	26 24	27	28 18	29	30 4	31	
32	33	34	35	36	37	38	
39	40 24	41	42 4	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2750630
OPR. CYCLE	2
DATE	06/25/75
ACC. BURNUP	764 MWD/TU
TH. POWER	2024.0 MW
FLOW RATE	$1.024 \cdot 10^4$ KG/SEC
SUBCOOLING	$5.279 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3	4	5	6	7	8
9	10 7	11	12 24	13	14 24	15	16
17	18	19	20	21	22	23	24
25	26 24	27	28 15	29	30 4	31	
32	33	34	35	36	37	38	
39	40 24	41	42 4	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2750717
OPR. CYCLE	2
DATE	07/17/75
ACC. BURNUP	1054 MWD/TU
TH. POWER	1936.0 MW
FLOW RATE	8940 KG/SEC
SUBCOOLING	J/KG
TIP SET 2	BOC 2

ROD PATTERN:

1	2	3	4	5	6	7	8
9	10 7	11	12 24	13	14 24	15	16
17	18	19	20	21	22	23	24
25	26 24	27	28 17	29	30 4	31	
32	33	34	35	36	37	38	
39	40 24	41	42 4	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2750724
OPR. CYCLE	2
DATE	07/24/75
ACC. BURNUP	1088 MWD/TU
TH. POWER	1948 MW
FLOW RATE	9211 KG/SEC
SUBCOOLING	$5.884 \cdot 10^4$ J/KG

QUAD CITIES - UNIT 2

ROD PATTERN:

1	2	3	4 1	5	6	7	8
9 5	10	11 24	12	13 7	14	15 24	16
17	18	19	20	21	22	23	24
25 24	26	27 14	28	29 24	30	31	
32	33	34	35 2	36	37	38	
39 16	40	41 24	42	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2750831
OPR. CYCLE	2
DATE	08/13/75
ACC. BURNUP	1354 MWD/TU
TH. POWER	1785.0 MW
FLOW RATE	8127 KG/SEC
SUBCOOLING	$6.581\cdot10^4$ J/KG

ROD PATTERN:

1	2	3	4 1	5	6	7	8
9 8	10	11 24	12	13 7	14	15 24	16
17	18	19	20	21	22	23	24
25 24	26	27 15	28	29 24	30	31	
32	33 2	34	35 2	36	37	38	
39 16	40	41 24	42	43	44		
45	46	47	48	49			
50	51	52					

←Control rod no. - PRESTO
←Rod insertion - nodes

DATASET NO.	QC 2751003
OPR. CYCLE	2
DATE	09/30/75
ACC. BURNUP	1752 MWD/TU
TH. POWER	2174 MW
FLOW RATE	$1.144\cdot10^4$ KG/SEC
SUBCOOLING	$5.502\cdot10^4$ J/KG

APPENDIX C

SELECTED RESULTS FROM MAJOR EVENTS AT QUAD CITIES 2

Cycle	Event no.	Date	Type of Event
1	6	7/5-1973	Rods moved at power
	13	21/9-1973	Rod withdrawal error
	23	5/4-1974	Rod swap
	25	15/4-1974	Fast start-up
	26	15/4-1974	Rods moved at power
	33	12/8-1974	Preconditioning
2	3	22/5-1975	Fast start-up
	4	22/5-1975	Rods moved at power

For each event:

Page	Table content
1	Assembly power after event
2	Axial power distribution after event
3	Peak ΔQ assemblywise
4	ΔQ-Q matrix
5	Failure probability map

PR-QC 2 0 PR-QC-12-03 REACTOR CONDITION 05-07-73:0357 AFTER
HYDRAULIC THROTTLING IDENT NUMBERS

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
3	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
6	1	1	1	1	1	1	1	1	1	1	1	1	1	2	
7	1	1	1	1	1	1	1	1	1	1	1	1	2		
8	1	1	1	1	1	1	1	1	1	1	1	1	2		
9	1	1	1	1	1	1	1	1	1	1	1	1	2		
10	1	1	1	1	1	1	1	1	1	1	1	2			
11	1	1	1	1	1	1	1	1	1	1	2				
12	1	1	1	1	1	1	1	1	1	2					
13	1	1	1	1	1	1	2	2	2						
14	1	1	1	1	1	2									
15	2	2	2	2	2										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 6

************ CORE AVERAGE EXPOSURE,MWD/TU = .28740E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	903	1248	1233	855	810	1068	1076	838	891	1247	1228	832	727	836	587
2	1250	1253	1232	1189	1112	947	949	1133	1218	1284	1258	1151	1028	899	598
3	1235	1232	1221	1188	1127	964	970	1142	1221	1275	1252	1144	1017	876	575
4	854	1186	1186	871	859	1155	1158	873	889	1224	1195	834	707	754	505
5	809	1109	1124	859	898	1255	1263	912	895	1186	1147	790	646	646	418
6	1067	946	963	1156	1257	1366	1374	1293	1226	1133	1072	1003	806	527	
7	1076	948	969	1160	1266	1374	1388	1302	1234	1127	1074	1024	717		
8	837	1131	1141	874	912	1292	1300	960	918	1167	1146	998	667		
9	891	1217	1220	890	896	1226	1232	918	891	1138	1090	889	569		
10	1251	1286	1278	1228	1189	1134	1128	1169	1140	1102	944	663			
11	1233	1261	1255	1200	1151	1073	1075	1149	1092	944	684				
12	835	1153	1146	836	792	1005	1026	999	890	664					
13	729	1028	1019	709	648	809	718	668	570						
14	838	903	879	757	648	529									
15	589	600	577	507	420										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 6

```
PR-QC    2  0 PR-QC-12-03  REACTOR CONDITION  05-07-73:0357 AFTER            PAGE  20
************ CORE AVERAGE EXPOSURE,MWD/TU =    .28740E+04


AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE


     NO. .0                      1.0                      2.0
BOTTOM
       1        2 1
       2             2   1
       3                 2     1
       4                    2        1
       5                     2            1
       6                      2              1
       7                     2                1
       8                    2                 1
       9                    2                1
      10                   2                1
      11                   2               1
      12                  2              1
      13                 2             1
      14                2            1
      15               2            1
      16              2           1
      17             2            1
      18            2           1
      19           2           1
      20           2         1
      21          2        1
      22        2       1
      23      2      1
      24   2   1
TOP
```

C-4

QUAD CITIES 2
CYCLE 1
EVENT NO.: 6

****
CASE TITLE = F1-QC-01-06 QUAD CITIES 2 STEP 6 . MAY 7 - 1973.

DQ.Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 6

Q * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	1463.	322478.	843321.	784494.	219639.	32004.	14143.	0.	0.
50.	0.	0.	9205.	5547.	19934.	16825.	7071.	4267.	7681.	0.
75.	0.	0.	0.	2438.	0.	0.	0.	0.	0.	0.
100.	0.	0.	0.	0.	3658.	2438.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	1707.	2926.	0.	0.	0.	0.
175.	0.	0.	0.	0.	1219.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 1
EVENT NO.: 6

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA2
****

CASE TITLE = F2-QC-01-30 QUAD CITIES - 2 , CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
7	0	0	0	0	0	0	0	0	0	0	0	0	0		
8	0	0	0	0	0	0	0	0	0	0	0	0	0		
9	0	0	0	0	0	0	0	0	0	0	0	0	0		
10	0	0	0	0	0	0	0	0	0	0	0	0			
11	0	0	0	0	0	0	0	0	0	0	0				
12	0	0	0	0	0	0	0	0	0	0					
13	0	0	0	0	0	0	0	0	0						
14	0	0	0	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 6

****

CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	1	2	4	5	6	4	0	0	0	1	0	0
2	1	1	1	2	2	4	6	7	10	10	4	19	19	4	1
3	1	1	1	2	3	5	6	8	12	14	11	27	26	4	1
4	1	2	2	3	9	16	15	8	9	0	0	19	13	0	0
5	2	2	3	9	21	42	42	21	11	2	5	22	10	0	0
6	4	4	5	16	42	169	171	42	15	19	22	11	0	0	
7	5	6	6	15	42	137	139	42	15	15	19	9	0		
8	6	0	2	8	21	41	41	21	11	10	3	2	1		
9	4	2	2	8	11	14	14	11	11	0	0	0	0		
10	13	14	16	21	20	18	15	6	5	5	2	1			
11	14	17	22	36	26	21	18	7	5	4	1				
12	6	4	8	39	42	24	17	9	5	0					
13	4	1	1	20	16	9	4	0	0						
14	2	4	2	0	0	1									
15	1	1	1	0	0										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 6

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1
****
CASE TITLE = F1-QC-01-13 QUAD CITIES 2 STEP 13 . SEPT 21 - 1973 RWE.

DQ.Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 13

NOTE:

Power distribution reported in main text.

Q / DQ	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	72847.	692384.	675071.	641909.	339791.	41026.	15301.	0.	0.
50.	0.	610.	37856.	70348.	49378.	16703.	3840.	0.	0.	0.
75.	0.	0.	18715.	22433.	18959.	11278.	0.	0.	0.	0.
100.	0.	0.	8778.	11460.	15362.	7132.	0.	0.	0.	0.
125.	0.	0.	6888.	12314.	5608.	5425.	0.	0.	0.	0.
150.	0.	0.	4450.	5730.	7498.	2682.	0.	0.	0.	0.
175.	0.	0.	5060.	7010.	5060.	0.	0.	0.	0.	0.
200.	0.	0.	1829.	13350.	2073.	0.	0.	0.	0.	0.
225.	0.	0.	914.	7437.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	3840.	0.	0.	0.	0.	0.	0.
275.	0.	0.	610.	2682.	0.	0.	0.	0.	0.	0.
300.	0.	0.	2682.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	3780.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 1
EVENT NO.: 13

****

CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	49	7	0	0	0	0	0	0	0	0	0	0	0	0	0
2	99	10	1	0	0	0	0	0	0	0	0	0	0	0	0
3	99	11	1	0	0	0	0	0	0	0	0	0	0	0	0
4	9	1	0	0	0	0	0	0	0	0	0	0	0	0	0
5	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
7	0	0	0	0	0	0	0	0	0	0	0	0	0		
8	0	0	0	0	0	0	0	0	0	0	0	0	0		
9	0	0	0	0	0	0	0	0	0	0	0	0	0		
10	0	0	0	0	0	0	0	0	0	0	0	0			
11	0	0	0	0	0	0	0	0	0	0	0				
12	0	0	0	0	0	0	0	0	0	0					
13	0	0	0	0	0	0	0	0	0						
14	0	0	0	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 13

****
CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	252	182	118	78	50	35	23	15	12	10	8	7	7	7	4
2	339	206	130	81	49	31	22	16	12	10	8	7	7	6	4
3	324	192	120	73	44	30	21	15	12	9	8	7	7	6	4
4	183	134	87	57	40	27	19	14	11	9	8	8	8	7	4
5	110	84	59	44	33	23	16	13	11	9	9	9	9	6	3
6	65	53	43	33	24	18	14	11	10	9	10	10	8	5	
7	38	36	31	24	18	15	13	11	9	9	9	9	6		
8	25	25	21	18	15	13	12	10	10	9	8	7	5		
9	18	17	15	14	12	11	11	11	10	9	6	5	3		
10	14	13	12	11	10	10	11	11	10	8	6	4			
11	11	10	10	9	9	9	10	10	9	7	4				
12	9	9	9	8	8	7	7	7	6	5					
13	9	8	7	8	7	5	4	4	4						
14	8	7	6	6	5	3									
15	5	5	4	4	3										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 13

************ CORE AVERAGE EXPOSURE,MWD/TU = .75319E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1049	1075	1111	1179	1198	1161	1134	1086	1071	1086	1110	1164	1142	1000	675
2	1065	811	823	1115	1138	917	910	1121	1098	826	825	1105	1123	983	661
3	1099	821	819	1085	1110	910	925	1162	1144	844	827	1077	1085	947	630
4	1171	1118	1039	1060	1076	1140	1182	1241	1233	1159	1108	1062	999	882	573
5	1183	1132	1106	1069	1085	1144	1191	1254	1257	1191	1129	1042	941	780	489
6	1128	881	876	1116	1129	927	945	1204	1212	995	931	1001	873	595	
7	1102	874	892	1156	1172	942	950	1188	1195	979	908	946	696		
8	1073	1113	1154	1229	1245	1209	1195	1196	1186	1153	1062	920	622		
9	1061	1097	1143	1223	1249	1218	1233	1185	1175	1156	1038	827	524		
10	1067	819	837	1141	1174	990	977	1144	1147	1065	891	615			
11	1093	820	621	1092	1114	927	905	1053	1023	888	639				
12	1162	1111	1081	1057	1038	1009	954	920	823	613					
13	1150	1134	1094	999	940	880	701	623	523						
14	1011	992	952	882	779	595									
15	677	663	631	572	489										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 23

```
PR-QC    2  0 PR-QC-10-11 REACTOR CONDITION,       04/05/74                  PAGE  20
************ CORE AVERAGE EXPOSURE,MWD/TU =    .75319E+04


AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE


      NO. .0                      1.0                      2.0
BOTTOM
        1       2   1
        2            2        1
        3                2           1
        4                  2            1
        5                   2           1
        6                   2           1
        7                   2          1
        8                   2          1
        9                   2          1
       10                  2           1
       11                  2           1
       12                  2           1
       13                 2           1
       14                 2           1
       15                 2          1
       16                 2          1
       17                2            1
       18                2            1
       19               2             1
       20              2             1
       21            2             1
       22          2          1
       23       2       1
       24    2    1
TOP
```

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QUAD CITIES 2
CYCLE 1
EVENT NO.: 23

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1
****

CASE TITLE = F1-QC-01-23 QUAD CITIES 2 STEP 23 APR 5 - 1974 ROD SEQ. B - A

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 23

Q / DQ	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	385999.	1092159.	1241328.	2493813.	153071.	8717.	0.	0.	0.
50.	0.	52608.	274991.	777606.	740725.	102840.	0.	0.	0.	0.
75.	0.	11582.	146304.	346801.	97719.	16154.	0.	0.	0.	0.
100.	0.	0.	25847.	158130.	6340.	2682.	0.	0.	0.	0.
125.	0.	0.	0.	6696.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 1
EVENT NO.: 23

****
CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

FAILURE PROBABILITY IN ASS. * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1	4	5	0	0	7	6	0	0	5	6	0	0	0	0
2	1	0	0	0	0	0	0	1	1	0	0	0	0	0	0
3	2	0	0	0	0	0	0	3	3	0	0	0	0	0	0
4	0	5	5	0	0	3	7	0	0	6	7	0	0	0	0
5	0	6	5	0	0	5	7	0	0	8	7	0	0	0	0
6	0	0	0	0	1	0	0	0	0	0	0	0	0	0	
7	0	0	0	2	2	0	0	0	0	0	0	0	0		
8	0	4	7	0	0	6	9	0	0	6	3	0	0		
9	0	5	6	0	0	9	9	0	0	10	3	0	0		
10	1	0	0	0	0	0	0	0	0	1	0	0			
11	2	0	0	0	0	0	0	0	0	0	0				
12	1	11	6	0	0	1	0	0	0	0					
13	2	16	9	0	0	0	0	0	0						
14	0	1	0	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 23

****
CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

MAXIMUM VALUE OF DO IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	59	85	93	45	45	89	80	44	44	83	93	43	34	15	11
2	87	32	21	72	72	31	29	83	83	27	23	80	83	16	12
3	99	30	23	65	65	28	33	98	97	30	27	62	64	19	13
4	48	94	78	33	33	79	96	49	47	92	82	38	35	25	15
5	31	88	78	30	30	79	91	32	32	85	85	40	36	25	15
6	18	37	37	80	80	34	35	14	19	43	45	24	24	17	
7	17	36	35	96	96	41	43	19	20	37	37	19	17		
8	35	78	94	51	51	93	85	36	40	88	82	30	18		
9	31	79	93	47	47	92	93	40	47	95	78	36	20		
10	83	29	23	51	52	38	38	71	77	55	48	30			
11	105	41	22	38	35	26	27	73	69	51	38				
12	76	109	87	38	32	77	73	49	46	36					
13	76	119	97	48	40	79	57	39	30						
14	59	60	44	34	31	29									
15	45	38	29	22	20										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 23

PR-QC 2 0 PR-QC-10-11 REACTOR CONDITION 04/15/74 PAGE 12

************ CORE AVERAGE EXPOSURE,MWD/TU = .75319E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1050	1072	1108	1179	1198	1157	1131	1089	1073	1077	1091	1128	1107	984	668
2	1065	815	827	1117	1139	920	913	1121	1097	827	822	1091	1108	974	656
3	1098	825	823	1086	1111	913	929	1160	1142	846	828	1073	1080	943	629
4	1170	1114	1086	1062	1077	1136	1178	1241	1232	1155	1105	1066	1004	883	574
5	1184	1129	1103	1071	1086	1141	1187	1254	1256	1187	1126	1045	945	782	490
6	1132	886	880	1117	1129	930	949	1205	1213	997	934	1003	875	597	
7	1107	880	897	1157	1173	946	953	1190	1197	982	911	947	698		
8	1079	1113	1153	1231	1246	1206	1193	1198	1187	1151	1059	920	623		
9	1068	1097	1142	1226	1251	1215	1200	1187	1176	1152	1034	826	525		
10	1067	824	844	1146	1179	996	982	1146	1147	1064	891	616			
11	1084	822	828	1099	1122	934	912	1055	1028	889	640				
12	1133	1101	1082	1068	1049	1015	957	922	825	615					
13	1122	1123	1094	1010	952	885	702	625	525						
14	1005	992	956	891	788	601									
15	677	665	636	578	494										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 25

************ CORE AVERAGE EXPOSURE,MWD/TU =   .75319E+04

AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE

```
      NO. .0                      1.0                      2.0
BOTTOM
        1       2   1
        2           2       1
        3               2         1
        4                 2           1
        5                   2           1
        6                   2           1
        7                    2         1
        8                    2          1
        9                    2          1
       10                   2           1
       11                   2           1
       12                  2           1
       13                  2          1
       14                 2           1
       15                 2           1
       16                 2          1
       17                 2           1
       18                2            1
       19                2            1
       20              2             1
       21            2            1
       22          2          1
       23       2       1
       24    2    1
TOP
```


QUAD CITIES 2
CYCLE 1
EVENT NO.: 25

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1
****

CASE TITLE = F1-QC-01-25 QUAD CITIES 2 STEP 25 APR 15 - 1974 REF. ROD WITHDRAWAL

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 25

Q / * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	671109.	876666.	819363.	268895.	8717.	0.	0.	0.	0.
50.	0.	133302.	839846.	2477363.	2134819.	16825.	0.	0.	0.	0.
75.	0.	0.	94366.	1051865.	750354.	92964.	0.	0.	0.	0.
100.	0.	0.	0.	28395.	39319.	2377.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 1
EVENT NO.: 25

****
CASE TITLE = F2-QC-01-00 CUAD CITIES - 2 , CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	3	5	5	2	2	6	6	3	3	7	9	5	4	1	0
2	2	0	0	1	1	0	1	2	2	0	1	2	2	1	0
3	3	1	0	1	1	0	1	3	3	1	1	2	2	1	0
4	4	5	5	2	3	4	6	3	4	5	7	4	3	0	0
5	3	6	5	2	2	6	7	3	3	7	6	4	1	0	0
6	1	0	1	1	2	0	1	1	2	2	1	1	0	0	
7	1	1	1	2	3	1	1	1	2	1	1	0	0		
8	4	5	7	2	4	6	8	3	4	5	3	1	0		
9	3	7	7	3	3	8	8	4	4	7	3	0	0		
10	1	0	0	1	1	1	1	1	2	2	1	0			
11	2	1	1	1	1	0	0	1	1	0	0				
12	8	13	10	4	2	1	1	0	0	0					
13	9	14	11	4	1	1	0	0	0						
14	2	3	2	1	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 25

****
CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

MAXIMUM VALUE OF DO IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	70	72	79	64	63	74	70	64	66	74	77	63	62	51	34
2	73	53	47	67	66	52	51	66	65	50	55	49	54	49	34
3	80	57	44	63	63	51	55	75	74	53	55	49	54	48	33
4	70	78	69	52	52	69	76	67	66	73	73	63	61	49	32
5	59	76	69	52	52	70	75	59	57	73	72	61	58	44	29
6	37	51	52	69	68	56	57	40	46	62	62	44	44	32	
7	41	51	51	76	76	60	62	43	47	58	56	39	32		
8	59	72	73	68	68	74	73	60	61	73	68	45	30		
9	59	73	72	59	60	75	76	63	67	76	68	47	29		
10	58	50	48	50	51	59	59	59	65	62	54	38			
11	63	53	53	41	39	51	50	61	64	54	43				
12	74	82	80	65	59	70	64	54	49	43					
13	82	88	84	70	63	68	52	42	34						
14	69	68	63	57	50	41									
15	47	46	43	39	33										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 25

************ CORE AVERAGE EXPOSURE,MWD/TU = .75319E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1049	1075	1111	1179	1198	1161	1134	1086	1071	1086	1110	1164	1142	1000	675
2	1065	811	823	1115	1138	917	910	1121	1098	826	825	1105	1123	983	661
3	1099	821	819	1085	1110	910	925	1162	1144	844	827	1077	1085	947	630
4	1171	1118	1069	1060	1076	1140	1182	1241	1233	1159	1108	1062	999	882	573
5	1183	1132	1105	1059	1085	1144	1131	1254	1257	1191	1129	1042	941	780	489
6	1128	881	876	1115	1129	927	945	1204	1212	995	531	1001	873	595	
7	1102	874	892	1156	1172	942	950	1188	1195	979	908	946	696		
8	1073	1113	1154	1229	1245	1209	1135	1196	1186	1153	1062	920	622		
9	1061	1097	1143	1223	1249	1218	1203	1185	1175	1156	1038	827	524		
10	1067	819	837	1141	1174	990	977	1144	1147	1065	891	615			
11	1093	820	621	1092	1114	927	935	1053	1023	888	639				
12	1162	1111	1081	1057	1038	1009	954	920	823	613					
13	1150	1134	1094	999	940	880	731	523	523						
14	1011	992	952	882	779	595									
15	677	663	631	572	488										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 26

```
PR-QC    2  0 PR-QC-10-11 REACTOR CONDITION        04/15/74                         PAGE  20
************ CORE AVERAGE EXPOSURE,MWD/TU =    .75319E+04


AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE


      NO. .0                       1.0                      2.0
BOTTOM
        1        2   1
        2             2       1
        3                 2          1
        4                   2            1
        5                    2           1
        6                    2           1
        7                    2          1
        8                    2          1
        9                    2          1
       10                   2           1
       11                   2           1
       12                   2           1
       13                  2           1
       14                  2           1
       15                  2          1
       16                  2          1
       17                 2            1
       18                 2            1
       19                2             1
       20               2             1
       21             2            1
       22           2          1
       23        2       1
       24     2    1
TOP
```

C-22

QUAD CITIES 2
CYCLE 1
EVENT NO.: 26

****

CASE TITLE = F1-QC-01-26 QUAD CITIES 2 STEP 26 APR 15 - 1974 ROD WITHDRAWN.

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 26

Q * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	239512.	492801.	680557.	1717975.	642823.	60899.	0.	0.	0.
50.	0.	0.	48463.	51572.	28346.	11095.	9632.	0.	0.	0.
75.	0.	0.	3292.	13777.	36942.	7742.	0.	0.	0.	0.
100.	0.	0.	0.	2743.	2743.	2073.	0.	0.	0.	0.
125.	0.	0.	0.	3292.	4939.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 1
EVENT NO.: 26

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA2
****
CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0
2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
7	0	0	0	0	0	0	0	0	0	0	0	0	0		
8	0	0	0	0	0	0	0	0	0	0	0	0	0		
9	0	0	0	0	0	0	0	0	0	0	0	0	0		
10	0	0	0	0	0	0	0	0	0	0	0	0			
11	0	0	0	0	0	0	0	0	0	0	0				
12	4	0	0	0	0	0	0	0	0	0					
13	4	0	0	0	0	0	0	0	0						
14	0	0	0	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 26

****

CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

MAXIMUM VALUE OF DO IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	1	1	0	0	1	3	6	15	32	66	78	69	54	30
2	1	0	0	0	0	0	0	5	10	21	43	67	71	48	28
3	0	0	0	0	0	0	0	4	8	12	23	34	38	31	18
4	0	1	1	0	1	1	2	3	5	9	13	18	2	1	1
5	0	1	1	0	0	1	1	1	3	6	7	4	0	0	0
6	0	0	0	0	0	0	0	0	0	1	1	0	0	0	
7	1	0	0	0	0	0	0	0	0	0	0	0	0		
8	5	6	5	2	1	1	1	0	0	1	1	0	0		
9	15	11	8	5	2	2	1	0	0	1	1	0	0		
10	30	21	11	6	3	0	0	0	0	0	0	0			
11	65	42	22	1[illegible]	5	1	0	0	0	0	0				
12	121	69	34	17	7	3	1	0	0	0					
13	122	73	39	20	8	4	2	0	0						
14	72	49	30	16	8	4									
15	35	28	19	11	5										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 26

************ CORE AVERAGE EXPOSURE,MWD/TU = .91738E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1103	1207	1260	1252	1285	1356	1376	1335	1299	1258	1151	962	827	719	487
2	780	1125	1192	1059	1089	1304	1335	1230	1192	1223	1092	694	578	660	470
3	765	1098	1163	1028	1059	1272	1306	1198	1169	1209	1092	699	586	663	465
4	1042	1118	1155	1157	1186	1263	1279	1243	1222	1221	1152	998	863	727	466
5	1025	1091	1114	1086	1110	1194	1204	1137	1130	1182	1165	1067	923	714	428
6	723	1027	1040	755	765	1078	1080	776	773	1091	1133	1047	874	574	
7	722	1028	1044	753	762	1068	1069	761	758	1064	1101	993	713		
8	1014	1109	1135	1087	1097	1169	1166	1091	1069	1107	1066	923	632		
9	1026	1127	1174	1159	1175	1221	1223	1174	1133	1091	997	815	528		
10	743	1079	1154	1066	1080	1214	1219	1133	1064	1001	855	602			
11	755	1084	1158	1064	1071	1185	1175	1070	973	852	618				
12	1067	1146	1177	1160	1142	1129	1065	960	828	608					
13	1103	1121	1129	1108	1061	973	774	669	548						
14	980	979	959	936	854	664									
15	671	667	651	609	534										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 33

```
PR-OC    2   0 REACTOR CONDITION 08/12/74
************ CORE AVERAGE EXPOSURE,MWD/TU =    .91738E+04


AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE


      NO. .0                      1.0                      2.0
BOTTOM
        1                 2             1
        2                          2                   1
        3                             2                     1
        4                             2                  1
        5                          2               1
        6                      2             1
        7                    2           1
        8                  2          1
        9                2          1
       10               2         1
       11               2        1
       12              2         1
       13              2         1
       14              2         1
       15              2         1
       16             2          1
       17             2          1
       18             2         1
       19            2          1
       20           2         1
       21          2        1
       22        2       1
       23      2     1
       24    2  1
TOP
```


QUAD CITIES 2
CYCLE 1
EVENT NO.: 33

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1
****
CASE TITLE = F1-QC-01-33 QUAD CITIES 2 STEP 33 AUG 12 - 1974.

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 33

Q * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	122469.	256581.	554065.	401300.	106314.	2682.	0.	0.	0.
50.	0.	72664.	243596.	312115.	234574.	321198.	43830.	0.	0.	0.
75.	0.	8352.	134478.	238902.	170566.	214213.	26518.	0.	0.	0.
100.	0.	0.	0.	76566.	126065.	59985.	4511.	0.	0.	0.
125.	0.	0.	0.	0.	36759.	34260.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 1
EVENT NO.: 33

****

CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	1	9	11	2	1	1	0	0	0	0	0
2	0	0	0	0	1	1	1	1	2	0	1	0	0	0	0
3	0	0	0	0	0	0	0	2	2	2	2	0	0	0	0
4	0	2	4	0	0	0	0	0	2	4	6	3	1	0	0
5	0	3	4	0	0	0	0	0	2	7	9	9	3	0	0
6	0	0	0	0	0	0	0	0	0	5	14	18	2	0	
7	0	0	0	0	0	0	0	0	0	4	11	8	0		
8	0	0	0	0	0	1	2	1	2	4	5	3	0		
9	0	0	0	0	0	1	2	3	4	4	2	1	0		
10	0	0	0	0	0	1	2	16	8	4	1	0			
11	0	0	0	0	0	0	2	11	4	1	0				
12	0	1	0	0	0	0	0	1	1	0					
13	0	0	0	0	0	0	0	0	0						
14	0	0	0	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 33

****

CASE TITLE = F2-QC-01-00 QUAD CITIES - 2 , CYCLE 1

MAXIMUM VALUE OF DO IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	43	47	52	58	78	107	107	87	73	71	65	52	39	24	13
2	32	47	50	64	67	82	84	70	65	73	71	51	34	13	13
3	40	59	59	56	59	65	68	67	65	89	89	68	53	31	22
4	52	102	103	61	55	62	65	66	84	110	110	94	79	62	37
5	61	101	102	61	54	60	64	66	86	112	114	108	93	71	42
6	39	56	57	40	41	58	62	56	73	101	114	106	88	62	
7	30	44	45	36	44	67	71	61	70	95	109	101	77		
8	39	44	46	45	59	103	104	81	80	90	97	93	68		
9	39	44	46	46	59	103	104	85	83	83	83	75	57		
10	31	46	48	51	52	72	79	101	101	79	69	54			
11	40	58	59	52	53	61	71	95	90	72	51				
12	61	103	102	63	55	57	61	70	70	48					
13	67	100	100	67	56	50	45	44	43						
14	54	60	59	51	44	36									
15	38	40	39	35	28										

QUAD CITIES 2
CYCLE 1
EVENT NO.: 33

************ CORE AVERAGE EXPOSURE,MWD/TU = .85931E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	946	900	989	1002	1042	961	1079	1105	1212	1064	1162	1164	1221	945	600
2	899	705	644	1042	938	750	704	1155	1047	834	812	1228	1186	1052	598
3	988	644	642	926	935	677	705	1039	1054	760	765	1096	1082	932	564
4	1006	1044	929	1135	1021	1107	1034	1299	1210	1263	1125	1286	1154	973	532
5	1042	938	936	1025	1033	1000	1081	1245	1239	1210	1169	1140	1027	763	449
6	962	749	677	1110	1006	817	783	1323	1266	1308	1113	1206	963	580	
7	1070	699	704	1035	1085	785	843	1257	1281	1255	1074	991	677		
8	1092	1143	1033	1302	1256	1343	1266	1485	1328	1403	1129	1019	582		
9	1193	1029	1050	1211	1266	1307	1326	1345	1392	1211	1038	778	472		
10	1042	819	755	1278	1256	1446	1383	1456	1224	1147	847	549			
11	1131	773	756	1132	1219	1226	1183	1168	1055	854	580				
12	1127	1201	1086	1291	1169	1262	1037	1054	790	560					
13	1191	1163	1071	1150	1039	982	700	601	481						
14	920	1030	917	964	759	579									
15	581	579	554	522	441										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 2

```
PR-QC    2  0 PR-QC-21-03 XE DYNAMICS CALC.CONDITION 5/21/75            PAGE  24
************ CORE AVERAGE EXPOSURE,MWD/TU =    .85931E+04

AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE

      NO. .0                      1.0                      2.0
BOTTOM
       1                 2              1
       2                         2                     1
       3                          2                        1
       4                        2                       1
       5                      2                     1
       6                    2                   1
       7                  2                 1
       8                 2               1
       9                2             1
      10               2            1
      11              2           1
      12             2          1
      13             2         1
      14            2         1
      15           2         1
      16           2        1
      17           2       1
      18          2       1
      19          2      1
      20         2      1
      21        2     1
      22       2    1
      23      2  1
      24    2 1
TOP
```


QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 2

****
CASE TITLE = F1-QC 02-03 QUAD CITIES 2 , CYCLE 2 STEP 3

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 3

Q * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	44562.	632277.	830153.	102657.	1829.	62728.	0.	0.	0.
50.	0.	2926.	77724.	381366.	377160.	15484.	38588.	0.	0.	0.
75.	0.	853.	40356.	186111.	428183.	199461.	29200.	6279.	0.	0.
100.	0.	0.	2743.	82174.	154290.	324917.	69129.	1829.	0.	0.
125.	0.	0.	0.	3536.	4511.	56571.	66020.	1829.	0.	0.
150.	0.	0.	510.	0.	1219.	4938.	5182.	0.	0.	0.
175.	0.	0.	0.	0.	1219.	4755.	1707.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 2

****
CASE TITLE = F2-QC-02-C0 QUAD CITIES - 2 , CYCLE 2, Q2.

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	1	1	2	1	6	2	7	2	23	3	4	0
2	0	0	0	0	1	0	0	0	5	0	2	0	6	0	0
3	0	0	0	0	1	0	1	7	7	1	2	9	11	3	0
4	1	0	1	0	2	0	6	0	11	0	20	0	2	0	0
5	1	1	1	3	3	4	5	9	24	31	18	17	5	0	0
6	2	0	0	0	3	0	2	0	26	0	32	0	2	0	
7	1	1	1	5	9	3	3	14	42	82	20	6	0		
8	4	0	6	0	9	0	13	0	45	0	20	0	0		
9	2	6	7	11	28	40	32	31	13	27	10	0	0		
10	6	0	2	0	36	0	66	0	26	3	1	0			
11	3	1	3	15	26	32	25	12	9	1	0				
12	9	0	9	0	20	0	7	0	1	0					
13	3	5	11	2	6	2	0	0	0						
14	5	0	3	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 2

****

CASE TITLE = F2-QC-02-00 QUAD CITIES - 2 , CYCLE 2, 02.

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	53	54	64	68	74	65	81	81	97	82	98	113	153	73	23
2	54	0	50	0	62	0	63	16	78	0	87	39	105	0	16
3	64	43	47	60	64	59	66	81	82	78	81	93	89	77	0
4	70	0	65	19	72	1	81	67	99	50	97	58	100	0	2
5	75	63	64	77	75	74	85	103	101	101	99	93	84	43	0
6	69	0	63	5	70	0	82	69	107	74	104	33	85	5	
7	81	66	66	79	89	83	93	106	108	167	97	72	47		
8	82	9	79	67	104	71	105	79	110	74	102	0	10		
9	97	78	83	98	104	108	111	114	117	101	65	49	0		
10	83	0	80	52	105	78	159	77	104	100	56	0			
11	98	87	81	93	100	104	101	93	87	59	0				
12	94	35	93	61	97	38	77	0	51	0					
13	152	104	91	102	86	86	43	3	0						
14	75	0	78	0	46	0									
15	19	43	15	0	0										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 2

PR-QC 2 0 PR-QC-21-03 XE DYNAMICS CALC.CONDITION 5/22/75 0020 PAGE 14

************ CORE AVERAGE EXPOSURE,MWD/TU = .85931E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1012	965	1047	1041	1067	972	1080	1100	1203	1053	1148	1148	1204	930	589
2	965	789	718	1083	960	758	705	1149	1040	826	803	1212	1170	1036	587
3	1047	719	709	950	953	683	705	1036	1050	757	760	1086	1070	920	555
4	1045	1084	962	1156	1033	1110	1033	1297	1210	1266	1126	1282	1146	963	525
5	1066	960	953	1035	1038	1000	1079	1246	1246	1226	1184	1145	1025	758	444
6	972	755	682	1109	1004	813	782	1326	1282	1369	1162	1226	967	579	
7	1069	698	701	1028	1076	779	838	1257	1295	1310	1121	1009	683		
8	1083	1131	1022	1287	1241	1327	1254	1477	1329	1420	1145	1029	586		
9	1178	1016	1036	1194	1248	1289	1310	1332	1385	1210	1039	778	473		
10	1026	805	743	1256	1236	1426	1365	1438	1212	1138	841	547			
11	1111	759	742	1112	1197	1207	1166	1151	1040	843	573				
12	1105	1177	1065	1265	1145	1239	1019	1035	776	550					
13	1167	1140	1049	1126	1018	962	686	558	471						
14	900	1007	897	943	742	566									
15	567	566	540	510	430										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 4

************ CORE AVERAGE EXPOSURE,MWD/TU =   .35931E+04

AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE

```
     NO.  .0                      1.0                       2.0
BOTTOM
        1                2               1
        2                         2                    1
        3                          2                       1
        4                         2                       1
        5                       2                     1
        6                     2                   1
        7                   2                 1
        8                 2                1
        9               2               1
       10              2              1
       11              2            1
       12             2           1
       13            2           1
       14            2         1
       15           2         1
       16           2         1
       17          2         1
       18          2        1
       19          2       1
       20         2      1
       21        2      1
       22       2      1
       23      2    1
       24    2  1
TOP
```


QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 4

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1
****
CASE TITLE = F1-QC 02-03 QUAD CITIES 2 , CYCLE 2 STEP 3

DQ.Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 3

Q / DQ	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	86380.	850511.	775777.	75225.	34199.	55047.	0.	0.	0.
50.	0.	1707.	146792.	710855.	467015.	54315.	37186.	1890.	0.	0.
75.	0.	0.	31943.	176967.	454884.	244389.	23774.	2499.	0.	0.
100.	0.	0.	0.	65288.	91501.	218298.	49560.	610.	0.	0.
125.	0.	0.	0.	2682.	9266.	42733.	30175.	0.	0.	0.
150.	0.	0.	0.	1463.	1219.	2926.	3048.	0.	0.	0.
175.	0.	0.	0.	0.	610.	1219.	1707.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 4

****
CASE TITLE = F2-QC-02-00 QUAD CITIES - 2 , CYCLE 2, Q4.

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1	1	0	2	1	2	1	5	2	5	2	16	2	3	0
2	1	0	0	0	1	0	0	0	3	0	1	0	4	0	0
3	0	0	0	1	1	0	0	3	2	0	0	3	4	2	0
4	3	0	1	0	2	0	3	0	2	0	2	0	1	0	0
5	2	2	2	4	4	3	2	2	1	1	1	1	1	0	0
6	4	0	1	0	4	0	0	0	1	0	4	0	0	0	
7	1	2	1	7	10	2	1	2	2	4	3	1	0		
8	8	0	10	0	10	0	7	0	3	0	1	0	0		
9	4	10	11	15	33	36	20	11	3	2	1	0	0		
10	11	0	4	0	44	0	59	0	9	1	0	0			
11	5	2	5	22	35	38	25	10	5	0	0				
12	17	0	14	0	29	0	8	0	1	0					
13	4	8	19	3	8	3	0	0	0						
14	9	0	5	0	1	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 4

****
CASE TITLE = F2-QC-02-00 QUAD CITIES - 2 , CYCLE 2, Q4.

MAXIMUM VALUE OF DO IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	65	64	73	76	81	69	82	78	92	76	91	106	146	68	20
2	64	0	58	0	68	0	62	10	69	0	76	29	95	0	15
3	75	52	56	67	68	59	60	69	65	55	57	73	73	65	0
4	80	0	73	25	75	0	70	47	68	8	57	24	73	0	4
5	86	72	72	83	77	70	70	73	55	50	49	45	47	27	0
6	79	0	70	9	78	0	63	30	52	3	66	0	46	10	
7	91	75	73	84	91	79	73	70	51	110	63	46	18		
8	92	17	86	71	107	67	91	52	64	0	52	0	6		
9	108	88	91	104	107	106	101	94	85	60	41	27	0		
10	93	0	88	57	109	78	156	66	84	74	31	0			
11	109	96	89	100	106	108	99	88	77	45	0				
12	105	44	103	69	103	42	79	0	47	0					
13	163	115	101	112	94	91	45	5	0						
14	84	0	87	0	51	1									
15	25	50	21	0	0										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 3 QUADRANT 4

```
************ CORE AVERAGE EXPOSURE.MWD/TU =    .85931E+04

AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE

     NO. .0                      1.0                      2.0
BOTTOM
       1                  2               1
       2                          2                      1
       3                           2                         1
       4                          2                        1
       5                       2                      1
       6                    2                   1
       7                  2                 1
       8                2               1
       9               2             1
      10              2           1
      11             2          1
      12            2          1
      13           2         1
      14           2        1
      15           2       1
      16          2        1
      17          2       1
      18          2      1
      19         2       1
      20        2       1
      21        2      1
      22       2     1
      23     2     1
      24   2   1
TOP
```


QUAD CITIES 2
CYCLE 2
EVENT NO.: 4

************ CORE AVERAGE EXPOSURE,MWD/TU = .85931E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1016	969	1050	1042	1066	969	1074	1093	1192	1042	1134	1133	1187	917	580
2	969	791	719	1084	959	755	700	1141	1032	818	793	1197	1154	1022	578
3	1051	721	709	960	952	680	701	1030	1044	752	754	1074	1058	907	547
4	1050	1087	964	1157	1032	1107	1029	1293	1208	1266	1124	1274	1135	952	518
5	1070	963	955	1036	1037	998	1076	1245	1252	1240	1196	1146	1020	751	438
6	976	757	683	1109	1003	811	780	1329	1296	1425	1207	1242	967	576	
7	1072	700	702	1028	1075	777	836	1259	1309	1360	1166	1024	688		
8	1087	1134	1023	1287	1239	1324	1251	1476	1333	1436	1158	1037	590		
9	1183	1019	1037	1194	1246	1285	1305	1328	1381	1210	1040	779	473		
10	1030	808	743	1257	1234	1419	1357	1431	1206	1133	838	546			
11	1116	762	744	1112	1196	1202	1159	1144	1034	838	570				
12	1111	1183	1068	1257	1145	1236	1014	1030	771	546					
13	1174	1145	1053	1129	1018	960	684	585	468						
14	905	1012	901	946	743	565									
15	570	568	542	511	430										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 4

****
CASE TITLE = F1-QC 02-04 QUAD CITIES 2 , CYCLE 2 STEP 4

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 4

Q * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	101011.	698419.	935370.	369235.	140086.	35052.	3597.	0.	0.
50.	0.	0.	0.	14691.	21763.	11400.	2987.	0.	0.	0.
75.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
100.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 2
EVENT NO.: 4 QUADRANT 2

****

CASE TITLE = F2-QC-02-00 QUAD CITIES - 2 , CYCLE 2, Q2.

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
7	0	0	0	0	0	0	0	0	0	0	0	0	0		
8	0	0	0	0	0	0	0	0	0	0	0	0	0		
9	0	0	0	0	0	0	0	0	0	0	0	0	0		
10	0	0	0	0	0	0	0	0	0	0	0	0			
11	0	0	0	0	0	0	0	0	0	0	0				
12	0	0	0	0	0	0	0	0	0	0					
13	0	0	0	0	0	0	0	0	0						
14	0	0	0	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 4 QUADRANT 2

****

CASE TITLE = F2-QC-02-00 QUAD CITIES - 2 , CYCLE 2, Q2.

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1	1	2	0	0	0	6	7	10	9	10	10	11	8	4
2	1	0	2	0	0	0	5	0	9	0	11	0	12	0	5
3	2	2	2	0	0	2	6	9	11	12	13	13	12	10	0
4	0	0	0	0	0	0	8	0	15	0	17	0	16	0	4
5	0	0	0	0	2	5	10	14	19	23	24	20	16	10	0
6	0	0	0	0	3	0	11	0	23	12	34	0	17	8	
7	1	0	1	3	4	6	11	16	23	38	33	23	12		
8	1	0	2	0	5	0	10	0	19	3	24	0	7		
9	2	1	2	3	5	6	9	11	16	17	17	13	0		
10	1	0	2	0	4	0	8	0	11	14	11	0			
11	1	2	2	3	4	5	6	7	8	8	0				
12	1	0	2	0	3	0	4	0	5	0					
13	1	1	1	2	2	3	1	0	0						
14	1	0	1	0	1	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 4 QUADRANT 2

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1
****

CASE TITLE = F1-QC 02-04 QUAD CITIES 2 , CYCLE 2 STEP 4

DQ.Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 4

Q / DQ	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	39258.	178552.	134844.	142585.	196840.	125882.	39624.	6888.	0.
50.	0.	0.	0.	0.	8656.	35540.	22799.	0.	0.	0.
75.	0.	0.	0.	0.	9266.	9876.	1829.	0.	0.	0.
100.	0.	0.	0.	0.	4145.	914.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	610.	914.	0.	0.	0.
150.	0.	0.	0.	0.	1463.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	1463.	0.	0.	0.	0.	0.
200.	0.	0.	0.	610.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

QUAD CITIES 2
CYCLE 2
EVENT NO.: 4 QUADRANT 4

****

CASE TITLE = F2-QC-02-00 QUAD CITIES - 2 , CYCLE 2, Q4.

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0	0	0	12	0	0	0	
7	0	0	0	0	0	0	0	0	0	91	7	0	0		
8	0	0	0	0	0	0	0	0	0	0	0	0	0		
9	0	0	0	0	0	0	0	0	0	0	0	0	0		
10	0	0	0	0	0	0	0	0	0	0	0	0			
11	0	0	0	0	0	0	0	0	0	0	0				
12	0	0	0	0	0	0	0	0	0	0					
13	0	0	0	0	0	0	0	0	0						
14	0	0	0	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 4 QUADRANT 4

****

CASE TITLE = F2-QC-02-00 QUAD CITIES - 2 , CYCLE 2, 04.

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	2	2	16	0	0	0	0	0	0	0	0	0	0	0	0
2	3	0	13	0	0	0	0	0	0	0	0	0	0	0	0
3	17	15	5	0	0	0	0	0	3	6	6	4	2	0	0
4	0	0	0	0	0	0	0	5	12	14	20	14	9	0	1
5	0	0	0	0	0	0	2	12	27	51	50	29	16	7	0
6	0	0	0	0	0	0	5	19	51	41	98	0	26	4	
7	0	0	0	0	0	0	4	19	50	182	90	53	14		
8	0	0	0	0	0	0	2	11	27	49	53	0	7		
9	1	0	0	0	0	0	0	4	13	21	23	17	0		
10	0	0	0	0	0	0	0	0	4	9	9	0			
11	1	0	0	0	0	0	0	0	0	2	0				
12	1	0	0	0	0	0	0	0	0	0					
13	2	1	0	1	0	0	0	0	0						
14	0	0	0	0	0	0									
15	0	0	0	0	0										

QUAD CITIES 2
CYCLE 2
EVENT NO.: 4 QUADRANT 4

APPENDIX D

SELECTED RESULTS FROM MAJOR EVENTS AT MAINE YANKEE

Cycle	Event no.	Date	Type of Event
1	7	1/10-1973	Startup Bank 5 withdrawn
	10	26/11-1973	Xe-transient
	12	1/1-1974	Fast startup
1A	1	10/10-1974	BOC 2A Fuel shuffled
	4	9/12-1974	Startup to 90.3% power

For each event:

Page	Table content
1	Assembly power after event
2	Axial power distribution after event
3	Peak ΔQ assemblywise
4	ΔQ-Q matrix
5	Failure probability map

PR-MA 1 1 MAINE YANKEE,STEP 7,SEPT.1 - OCT.12-73
************ CORE AVERAGE EXPOSURE,MWD/TU = .40996E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	1133	1051	1046	1122	1120	1040	1040	1119	1123	1050	1059	1148	1149	1031	940	952	584
2		1126	1124	1044	1040	1115	1115	1040	1045	1128	1134	1065	1061	1275	1172	948	578
3			1122	1041	1038	1113	1113	1037	1040	1122	1127	1055	1055	1261	1136	937	568
4				1114	1112	1035	1034	1111	1111	1034	1036	1117	1113	1176	1005	698	385
5					1110	1032	1031	1106	1104	1025	1020	1087	1058	1063	756		
6						1106	1104	1026	1023	1094	1080	979	920	978	658		
7							1101	1024	1021	1089	1072	951	871	901	575		
8								1099	1095	1009	981	1146	1026	862	537		
9									1092	996	956	1089	929	642	358		
10										1193	1115	1008	712				
11											986	730	448				

MAINE YANKEE
CYCLE 1
EVENT NO.: 7

```
************ CORE AVERAGE EXPOSURE,MWD/TU =    .45437E+04

AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE

     NO. .0                      1.0                      2.0
BOTTOM
      1           1
      2                1
      3                    1
      4                        1
      5                          1
      6                            1
      7                              1
      8                              1
      9                               1
     10                                1
     11                                1
     12                                1
     13                                1
     14                                1
     15                                1
     16                                1
     17                                1
     18                               1
     19                             1
     20                           1
     21                        1
     22                   1
     23              1
     24       1
TOP
```

MAINE YANKEE
CYCLE 1
EVENT NO.: 7

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1
****
CASE TITLE = F1-MY 01-07 MAINEE YANKEE - CYCLE1 - STEP 7

DQ.Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 7

***** * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
25.	0.	0.	921515.	915886.	35256.	0.	0.	0.	0.	0.
50.	0.	0.	1384651.	4511009.	105768.	0.	0.	0.	0.	0.
75.	0.	0.	87810.	327220.	45172.	0.	0.	0.	0.	0.
100.	0.	0.	0.	4914.	0.	0.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

MAINE YANKEE
CYCLE 1
EVENT NO.: 7

****
CASE TITLE = F2-MY-01 MAINE YANKEE CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	5	3	1	0	0	0	1	1	1	2	4	6	7	3	1	1	0
2		1	1	1	0	1	1	1	1	2	3	4	4	7	4	1	0
3			1	1	1	1	1	1	1	2	2	3	3	6	3	1	0
4				1	1	1	1	1	1	2	2	2	2	4	1	0	0
5					1	1	1	1	1	2	2	2	1	2	0		
6						1	1	1	2	2	1	1	1	1	0		
7							1	1	2	2	2	1	1	1	0		
8								2	2	2	2	3	2	0	0		
9									2	2	2	3	1	0	0		
10										9	6	2	0				
11											4	0	0				

MAINE YANKEE
CYCLE 1
EVENT NO.: 7

CASE TITLE = F2-MY-01 MAINE YANKEE CYCLE 1

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	57	48	35	28	25	26	29	31	35	43	51	72	73	56	47	45	0
2		38	32	29	26	26	28	33	37	40	46	54	57	62	54	44	0
3			30	28	27	27	28	33	37	39	43	49	50	57	50	39	0
4				27	27	29	31	32	35	40	43	44	44	52	45	26	0
5					27	30	32	33	36	40	42	41	41	46	33		
6						30	32	36	39	39	40	41	39	42	26		
7							34	38	41	41	41	41	38	38	0		
8								39	41	45	45	52	45	34	0		
9									44	49	49	52	43	3	0		
10										79	74	48	24				
11											69	22	0				

MAINE YANKEE
CYCLE 1
EVENT NO.: 7

************ CORE AVERAGE EXPOSURE,MWD/TU = .56607E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	651	832	949	1078	1118	1065	1076	1144	1128	1026	958	871	873	937	929	969	612
2		976	1045	1025	1054	1141	1150	1078	1067	1115	1078	965	967	1222	1162	963	606
3			1082	1043	1064	1146	1155	1083	1076	1133	1116	1035	1038	1250	1142	955	593
4				1130	1146	1085	1089	1159	1153	1072	1062	1127	1118	1193	1040	723	408
5					1156	1090	1092	1161	1153	1073	1063	1120	1089	1107	806		
6						1164	1163	1086	1076	1139	1122	1028	977	1045	713		
7							1157	1078	1067	1121	1104	1000	932	967	628		
8								1136	1114	1021	996	1176	1067	908	580		
9									1079	953	915	1067	939	671	388		
10										1006	920	912	685				
11											757	647	421				

MAINE YANKEE
CYCLE 1
EVENT NO.: 10

************ CORE AVERAGE EXPOSURE,MWD/TU =   .56607E+04

AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE

```
     NO. .0                  1.0                2.0
BOTTOM
      1          1
      2               1
      3                  1
      4                     1
      5                      1
      6                       1
      7                        1
      8                        1
      9                        1
     10                         1
     11                         1
     12                         1
     13                         1
     14                         1
     15                        1
     16                        1
     17                        1
     18                       1
     19                      1
     20                     1
     21                   1
     22                1
     23            1
     24       1
TOP
```

MAINE YANKEE
CYCLE 1
EVENT NO.: 10

****
CASE TITLE = F1-MY 01-10 MAINEE YANKEE - CYCLE1 - STEP 10

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 1

*****	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.
*										
DQ										
*										
25.	0.	2534.	1144499.	2559147.	105870.	0.	0.	0.	0.	0.
50.	0.	0.	534019.	2354969.	100259.	0.	0.	0.	0.	0.
75.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
100.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

MAINE YANKEE
CYCLE 1
EVENT NO.: 10

****
CASE TITLE = F2-MY-01 MAINE YANKEE CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	0	0	0	0	1	1	1	1	1	0	0	0	0	0	0	0	0
2		0	0	1	1	1	1	1	1	1	0	0	0	1	1	0	0
3			1	1	1	1	1	1	1	1	1	0	0	1	1	0	0
4				1	1	1	1	1	1	1	1	1	0	1	0	0	0
5					1	1	1	1	1	1	1	1	0	1	0		
6						1	1	1	1	1	1	0	0	0	0		
7							1	1	1	1	0	0	0	0	0		
8								1	1	0	0	1	0	0	0		
9									0	0	0	0	0	0	0		
10										0	0	0	0				
11											0	0	0				

MAINE YANKEE
CYCLE 1
EVENT NO.: 10

****
CASE TITLE = F2-MY-01 MAINE YANKEE CYCLE 1

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	17	28	32	33	34	36	37	35	34	33	30	25	24	28	28	30	15
2		30	32	35	36	36	36	36	36	33	31	30	29	37	35	30	15
3			33	36	36	36	36	37	36	34	33	32	32	38	35	28	0
4				35	36	37	37	36	35	35	35	33	32	38	34	21	0
5					36	37	37	36	35	36	35	33	32	36	27		
6						36	36	37	36	34	33	33	32	34	22		
7							36	36	35	33	32	32	30	31	16		
8								34	33	33	32	37	34	27	0		
9									31	29	28	33	29	17	0		
10										29	27	26	19				
11											22	15	0				

MAINE YANKEE
CYCLE 1
EVENT NO.: 10

************ CORE AVERAGE EXPOSURE,MWD/TU = .69903E+04

CHANNEL POWER DISTRIBUTION.AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	758	928	1034	1151	1180	1124	1120	1174	1150	1048	981	898	890	932	909	929	579
2		1067	1128	1101	1118	1190	1187	1110	1092	1127	1085	976	969	1196	1116	911	569
3			1159	1111	1122	1192	1188	1111	1096	1140	1114	1031	1024	1207	1083	892	548
4				1187	1193	1127	1121	1176	1160	1078	1061	1108	1081	1139	987	671	375
5					1194	1124	1117	1172	1156	1075	1057	1096	1049	1057	769		
6						1185	1175	1098	1083	1130	1105	1010	954	1004	678		
7							1164	1086	1069	1110	1085	983	910	926	595		
8								1133	1107	1015	936	1149	1026	856	540		
9									1068	952	909	1034	894	627	359		
10										1009	910	876	646				
11											744	617	397				

MAINE YANKEE
CYCLE 1
EVENT NO.: 12

```
FR-MA    1   1 MAINE YANKEE,STEP 10.DEC.31 - JAN.31-74
************ CORE AVERAGE EXPOSURE,MWD/TU =    .69903E+04


AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE


       NO.  .0                           1.0                           2.0
BOTTOM
         1                1
         2                       1
         3                             1
         4                                  1
         5                                   1
         6                                   1
         7                                   1
         8                                   1
         9                                   1
        10                                   1
        11                                   1
        12                                   1
        13                                   1
        14                                   1
        15                                    1
        16                                    1
        17                                    1
        18                                    1
        19                                   1
        20                                  1
        21                             1
        22                        1
        23                 1
        24         1
TOP
```

D-13

MAINE YANKEE
CYCLE 1
EVENT NO.: 12

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1

****

CASE TITLE = F1-MY 01-12 MAINEE YANKEE - CYCLE1 - STEP 12

MAINE YANKEE
CYCLE 1
EVENT NO.: 12

DO.Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DO AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 12

***** * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.	550.	600.	650.
25.	0.	89572.	2535809.	5823304.	438056.	0.	0.	0.	0.	0.	0.	0.	0.
50.	0.	1102.	277421.	828517.	76021.	0.	0.	0.	0.	0.	0.	0.	0.
75.	0.	0.	5941.	2204.	0.	0.	0.	0.	0.	0.	0.	0.	0.
100.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.	0.	0.	0.	0.	0.	0.

****

CASE TITLE = F2-MY-01 MAINE YANKEE CYCLE 1

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0
2		1	1	1	1	1	1	1	1	1	1	1	1	2	1	0	0
3			1	1	1	1	1	1	1	1	1	1	1	2	1	0	0
4				1	1	1	1	1	1	1	1	1	1	1	1	0	0
5					1	1	1	1	1	1	1	1	1	1	0		
6						1	1	1	1	1	1	1	1	1	0		
7							1	1	1	1	1	1	1	1	0		
8								1	1	1	1	2	1	0	0		
9									1	1	1	1	0	0	0		
10										2	1	0	0				
11											0	0	0				

MAINE YANKEE
CYCLE 1
EVENT NO.: 12

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA2
****
CASE TITLE = F2-MY-01   MAINE YANKEE   CYCLE 1

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	51	43	33	29	27	28	28	27	28	32	37	61	60	38	31	30	18
2		33	30	30	29	26	26	28	29	29	31	38	38	39	34	28	15
3			29	29	28	26	26	28	29	28	29	32	32	35	31	23	9
4				27	26	28	28	26	27	29	30	28	27	32	29	18	0
5					26	27	28	26	27	29	29	27	26	30	23		
6						26	26	28	28	27	27	28	28	29	20		
7							27	28	29	28	28	29	27	27	17		
8								28	29	31	31	34	30	22	8		
9									30	35	34	33	28	16	0		
10										71	67	32	21				
11											54	19	0				

MAINE YANKEE
CYCLE 1
EVENT NO.: 12

DP-4A 1 1 MAINE YANKEE • CYCLE 1A • STEP 1 • OCT.27 - NOV.7 -74 PAGE 11

************ CORE AVERAGE EXPOSURE,MWD/TU = .65046E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	826	659	927	1070	1108	1039	1054	1137	1143	1077	1057	1013	981	913	801	769	469
2		879	938	971	1027	1145	1141	1074	1081	1157	1172	1168	1107	1092	941	734	456
3			993	989	1042	1169	1163	1098	1106	1176	1191	1190	1120	1065	892	708	428
4				1153	1179	1105	1116	1201	1203	1127	1116	1170	1106	1011	820	534	296
5					1198	1164	1177	1241	1238	1177	1155	1165	1074	957	665		
6						1386	1400	1205	1195	1363	1343	1064	938	964	580		
7							1410	1209	1191	1349	1319	1020	884	886	505		
8								1252	1224	1138	1056	1105	966	830	472		
9									1181	1043	948	981	826	576	292		
10										1113	1004	870	607				
11											869	620	384				

MAINE YANKEE
CYCLE 1A
EVENT NO.: 1

```
************ CORE AVERAGE EXPOSURE,MWD/TU =    .68046E+04


AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE


     NO.  .0                  1.0                 2.0
BOTTOM
       1           1
       2                1
       3                   1
       4                    1
       5                     1
       6                      1
       7                      1
       8                      1
       9                      1
      10                     1
      11                     1
      12                     1
      13                     1
      14                      1
      15                      1
      16                      1
      17                      1
      18                      1
      19                      1
      20                      1
      21                    1
      22                 1
      23            1
      24      1
TOP
```

MAINE YANKEE
CYCLE 1A
EVENT NO.: 1

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA1
****

CASE TITLE = F1-MY 02-01 MAINE YANKEE CYCLE 1A STEP 1.

MAINE YANKEE
CYCLE 1A
EVENT NO.: 1

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 1

Q / * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.	550.	600.	650.
25.	0.	189281.	600014.	2258810.	4407.	0.	0.	0.	0.	0.	0.	0.	0.
50.	13441.	0.	99819.	18730.	0.	0.	0.	0.	0.	0.	0.	0.	0.
75.	16857.	94200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
100.	0.	108192.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

CASE TITLE = F2-MY-01 MAINE YANKEE CYCLE 1A

FAILURE PROBABILITY IN ASS, * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0
2		0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
3			0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4				0	0	0	0	0	0	0	0	0	0	0	0	0	0
5					0	0	0	0	0	0	0	0	0	0	0		
6						0	0	0	0	0	0	0	0	5	0		
7							0	0	0	0	0	0	0	1	0		
8								0	0	0	0	0	0	0	0		
9									0	104	80	0	0	0	0		
10										1	1	0	0				
11											1	0	0				

MAINE YANKEE
CYCLE 1A
EVENT NO.: 1

****

CASE TITLE = F2-MY-01 MAINE YANKEE CYCLE 1A

MAXIMUM VALUE OF DQ IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	0	0	0	0	0	4	6	0	0	10	9	23	14	9	0	0	0
2		0	0	10	16	0	0	7	9	0	0	0	0	4	0	0	0
3			8	8	13	0	0	0	11	0	0	0	0	2	0	0	0
4				0	0	9	9	0	0	12	15	0	0	4	0	0	0
5					0	12	13	0	0	15	18	0	0	4	0		
6						0	0	14	14	0	0	14	7	38	0		
7							0	14	13	0	0	10	3	24	0		
8								0	0	9	2	2	1	23	0		
9									0	86	80	0	0	6	0		
10										28	26	5	0				
11											28	4	0				

MAINE YANKEE
CYCLE 1A
EVENT NO.: 1

PR-MA 1 1 MAINE YANKEE , CYCLE 1A , STEP 3 , DEC.8 - JAN.1 -75 PAGE 11
************ CORE AVERAGE EXPOSURE,MWD/TU = .80282E+04

CHANNEL POWER DISTRIBUTION,AVERAGE CH.POW.= 1000

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	632	866	937	1104	1146	1054	1067	1167	1175	1093	1085	1048	1018	946	827	790	484
2		887	953	992	1048	1180	1168	1083	1091	1189	1209	1215	1152	1114	961	749	469
3			1012	1008	1058	1198	1184	1099	1108	1201	1222	1225	1155	1079	905	720	438
4				1195	1203	1101	1107	1205	1211	1121	1117	1194	1132	1019	830	543	303
5					1209	1143	1149	1234	1231	1150	1137	1180	1093	959	673		
6						1356	1362	1163	1154	1327	1317	1049	937	967	583		
7							1364	1159	1142	1302	1280	996	876	878	503		
8								1225	1190	1085	1011	1067	939	811	464		
9									1144	995	907	938	796	560	287		
10										1058	954	829	582				
11											828	592	369				

MAINE YANKEE
CYCLE 1A
EVENT NO.: 4

```
PR-MA   1  1 MAINE YANKEE , CYCLE 1A , STEP 3 , DEC.8  - JAN.1 -75          PAGE  14
************ CORE AVERAGE EXPOSURE,MWD/TU =    .80282E+04

AVERAGE AXIAL POW.DISTR.PER THROTL.TYPE

      NO. .0                       1.0                      2.0
BOTTOM
         1              1
         2                     1
         3                         1
         4                           1
         5                            1
         6                            1
         7                            1
         8                           1
         9                           1
        10                           1
        11                           1
        12                           1
        13                           1
        14                           1
        15                           1
        16                           1
        17                            1
        18                            1
        19                            1
        20                            1
        21                          1
        22                       1
        23                 1
        24         1
TOP
```

MAINE YANKEE
CYCLE 1A
EVENT NO.: 4

****
CASE TITLE = F1-MY 02-04 MAINE YANKEE CYCLE 1A STEP 4.

MAINE YANKEE
CYCLE 1A
EVENT NO.: 4

DQ,Q - MATRIX GIVING CM OF ROD PER INTERVAL IN DQ AND Q
EXTRAPOLATED TO FULL CORE.
UPPER INTERVAL BOUNDARIES GIVEN IN W/CM.
TIME STEP NO = 4

Q * DQ *	50.	100.	150.	200.	250.	300.	350.	400.	450.	500.	550.	600.	650.
25.	4958.	1170390.	2179153.	3687891.	2534358.	68309.	0.	0.	0.	0.	0.	0.	0.
50.	0.	0.	83072.	307168.	213740.	46274.	0.	0.	0.	0.	0.	0.	0.
75.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
100.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
125.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
150.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
175.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
200.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
225.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
250.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
275.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
300.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
325.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
350.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

****

CASE TITLE = F2-MY-C1   MAINE YANKEE CYCLE 1A

FAILURE PROBABILITY IN ASS. * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	2	2	2	0	0	3	3	0	0	3	3	3	3	2	2	1	0
2		2	2	3	3	0	0	2	3	0	0	0	0	3	2	1	0
3			3	3	3	0	0	0	2	0	0	0	0	3	2	1	0
4				0	0	2	2	0	0	2	3	0	0	2	1	0	0
5					0	2	2	0	0	2	2	0	0	2	1		
6						0	0	2	2	0	0	2	2	2	0		
7							0	2	2	0	0	2	1	2	0		
8								0	0	2	1	2	1	0	0		
9									0	4	3	1	1	0	0		
10										2	1	1	0				
11											1	0	0				

MAINE YANKEE
CYCLE 1A
EVENT NO.: 4

****
CASE TITLE = F2-MY-C1 MAINE YANKEE CYCLE 1A

FAILURE PROBABILITY IN ASS. * 100.0

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	2	2	2	0	0	3	3	0	0	3	3	3	3	2	2	1	0
2		2	2	3	3	0	0	2	3	0	0	0	0	3	2	1	0
3			3	3	3	0	0	0	2	0	0	0	0	3	2	1	0
4				0	0	2	2	0	0	2	3	0	0	2	1	0	0
5					0	2	2	0	0	2	2	0	0	2	1		
6						0	0	2	2	0	0	2	2	2	0		
7							0	2	2	0	0	2	1	2	0		
8								0	0	2	1	2	1	0	0		
9									0	4	3	1	1	0	0		
10										2	1	1	0				
11											1	0	0				

MAINE YANKEE
CYCLE 1A
EVENT NO.: 4

SCP - FUEL DUTY CYCLE ANALYSIS REPORT - PROGRAM FDCA2
****
CASE TITLE = F2-MY-C1   MAINE YANKEE CYCLE 1A

MAXIMUM VALUE OF DO IN ASSEMBLY

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	22	22	23	19	24	25	25	25	25	25	26	32	28	26	24	23	15
2		23	24	27	27	26	25	24	25	26	26	27	25	28	25	21	14
3			26	27	27	26	25	0	24	25	26	26	24	26	24	19	12
4				27	26	24	24	25	25	24	24	25	21	26	23	15	0
5					26	24	23	25	25	23	24	25	16	24	20		
6						27	26	23	23	26	27	24	24	27	12		
7							26	22	22	25	25	23	22	24	0		
8								24	23	21	21	24	22	20	11		
9									21	28	27	21	19	15	0		
10										29	27	18	14				
11											26	15	9				

MAINE YANKEE
CYCLE 1A
EVENT NO.: 4

APPENDIX E

DESIGN AND CORE OPERATIONS QUESTIONNAIRE

FOR

FUEL DUTY CYCLE ANALYSIS

Section 1. Fuel Type and Location Definition by Operating Cycle.

Section 2. Fuel Failure and Inspection Information Summary by Cycle.

Section 3. Technical Information for Simulation of Core Performance

Section 4. Operating History by Cycle.

Section 5. Manufacturing and Design Data for Fuel Types Employed.

Section 6. Figures: This section is reserved for insertion of core maps, plots and other Figures requested in Sections 1 to 5.

Section 7. Tables: This section contains blank tables set up for tabular information requested in sections 1 through 5.

Section 8. Exhibits: This section contains samples illustrating the type of material being requested in certain questions of Sections 1 to 5.

SECTION 1 - FUEL TYPE AND LOCATION DEFINITION BY OPERATING CYCLE

1.1 GENERAL INFORMATION

1.2 FUEL IDENTIFICATION

1.3 FUEL LOCATION

1.4 ADDITIONAL OR SUPPLEMENTARY DATA

NOTE: The objectives of this group of data are to:

- Identify which types of fuel have been employed by some characteristic number which can be used to locate detailed data in section v
- Provide means to trace the location history of multi-cycle fuel which has been shuffled. This is accomplished by some form of science numbering system
- Provide cross section showing initial enrichment distributions for new fuel in the various charges

1.1 GENERAL INFORMATION

1.1.1 NAME OF OPERATING UTILITY ....

(a) Address ....

(b) Telephone No. ....

(c) Name of Nuclear Station ....

1.1.2 NAME OF PERSON PREPARING RESPONSE ....

(a) Office Telephone No. ....

(b) Position ....

(c) Date of Mailing Response ....

1.1.3 TYPE OF REACTOR ....

(a) NSSS System .... (Firm)

(b) Nominal MWe ....

(c) NSSS Suppliers Type Identified .... (e.g. BWR-4)

1.1.4 HISTORICAL DATA

(a) Date of First Full Power Operation ....

(b) No. of Completed Cycles ....

(c) No. of Current Cycle ....

(d) Previous Cycles as Below ....

Cycle No.	*Date Start*	*Date End*
1	....	....
2	....	....
3	....	....

1.2 FUEL IDENTIFICATION

1.2.1 PLEASE IDENTIFY TYPES AND AMOUNTS OF FUEL THAT HAVE BEEN EMPLOYED BY FILLING IN TABLE 1.2.1. PLEASE PROVIDE ASSEMBLY DRAWINGS SIMILAR TO EXHIBIT 1.2.1 IF POSSIBLE. (INSERT AS FIGURES 1.2.1.1 TO 1.2.1.X)

1.3 FUEL LOCATION

1.3.1 PLEASE PROVIDE A CORE MAP FOR EACH CYCLE SHOWING WHICH FUEL ASSEMBLY IS LOCATED WHERE. USE SEQUENCE NUMBERS AS DEFINED IN COLUMN VII OF TABLE 1.2.1 AN EXAMPLE IS GIVEN IN EXHIBIT 1.3.1. ATTACH THESE MAPS AS FIGURES 1.3.1.(1) → 1.3.1.(x). (Please provide additional core maps if mid cycle changes in fuel location have occurred).

1.3.2 PLEASE PROVIDE A CORE MAP SHOWING CONTROL ROD CONFIGURATION UPON WHICH CONTROL ROD POSITION DESCRIPTIONS ARE BASED. EXHIBIT 1.3.2.1 GIVES AN EXAMPLE. INCLUDE THIS MAP AS FIGURE 1.3.2 OF FINISHED QUESTIONNAIRE.

1.4 ADDITIONAL OR SUPPLEMENTARY INFORMATION

1.4.1 IF EFFORT TO PREPARE THIS QUESTIONNAIRE COMPLETELY IS MORE EXTENSIVE THAN CAN BE ALLOCATED BY YOU AT THIS TIME, COULD WE BE ALLOWED TO VISIT YOUR FACILITY TO OBTAIN ADDITIONAL CRITICAL DATA VIA DISCUSSIONS AND INTERVIEWS WITH YOUR STAFF?

....

1.4.2 HAVE YOUR OWN EFFORTS TO LIMIT ECONOMIC RISKS, ASSOCIATED WITH UNPREDICTABLE FUEL FAILURES, IDENTIFIED USEFUL PUBLICATIONS OR STUDIES WITH A SCOPE OR OBJECTIVES SIMILAR TO THIS ONE? WOULD YOU RECOMMEND ANY OF THESE AS NOTABLY SUCCESSFUL OR AS A SOURCE OF IMPORTANT REFERENCE MATERIAL FOR THE CURRENT STUDY ....
.... ?

1.4.3 THERE WILL PROBABLY BE SOME DESIGN AND INSPECTION INFORMATION, REQUESTED HEREIN, THAT WOULD REQUIRE SOME BREACH OF CONFIDENTIALITY WITH YOUR FUEL SUPPLIER IF SUPPLIED DIRECTLY BY YOUR ORGANIZATION. WOULD YOU HAVE RESERVATIONS ABOUT OUR PURSUING ANY OF THESE QUESTIONS DIRECTLY WITH SUPPLIERS? .... IF SO, COULD THESE RESERVATIONS BE DEALT WITH BY SUBMITTING TO YOU FOR PRIOR APPROVAL CORRESPONDENCE DIRECTED AT THE SUPPLIER? ....

SECTION 2 - FUEL FAILURE AND INSPECTION INFORMATION SUMMARY BY CYCLE

2.1 MACROSCOPIC INFORMATION CHARACTERIZING PROGRAM

2.2 TABULAR INFORMATION ON FUEL INSPECTIONS

NOTE: If complex answers seem desirable, use the footnotes

2.1 MACROSCOPIC INFORMATION - PROGRAM CHARACTERIZATION

(These answers need not be precise, approximations are sufficient)

2.1.1 HOW MANY FUEL BUNDLES HAVE BEEN INSERTED IN THIS REACTOR? ....

2.1.2 HOW MANY FUEL BUNDLES HAVE BEEN REMOVED PERMANENTLY? ....

2.1.3 HOW MANY ASSEMBLIES HAVE CONTAINED AT LEAST ONE LEAKY ROD (BASED ON SIPPING)? ....

2.1.4 HOW MANY LEAKY ASSEMBLIES (BASED ON SIPPING) HAVE BEEN COMPLETELY DISASSEMBLED AND SUBJECTED TO EXAMINATION IN HOT LABORATORIES? ....

2.1.5 HOW MANY LEAKY ASSEMBLIES (BASED ON SIPPING) HAVE BEEN DISASSEMBLED AND SUBJECTED TO EDDY CURRENT AND ULTRASONIC EXAMINATION IN BASIN? ....

2.1.6 HOW MANY LEAKY ASSEMBLIES HAVE BEEN EXAMINED WITH VIDEO OR PHOTOGRAPHIC RECORDS? ....

2.1.7 HOW MANY LEAKY ASSEMBLIES HAVE BEEN OPTICALLY EXAMINED WITHOUT VIDEO OR PHOTOGRAPHIC RECORDS? ....

2.1.8 HOW MANY NON LEAKING ASSEMBLIES (BASED ON SIPPING),

- (a) Have been completely disassembled and subjected to metallographic PIE? ....
- (b) Have been disassembled and subjected to ultrasonic or eddy current examination in basin? ....
- (c) Have been disassembled and subjected to visual exam. or profilometry in basin? ....
- (d) Have been examined superficially with video records kept? ....
- (e) Have been optically examined with no records kept? ....

2.1.9 IS THERE A REPORT(S) SUMMARIZING THE INSPECTION RESULTS FOR THIS REACTOR? IF SO, IDENTIFY: ....

CAN ANY OF THESE BE INCLOSED WITH RESPONSE? ....

2.1.10 IS THE FUEL INSPECTION PROGRAM:

- (a) Conducted by the fuel supplier under contract ...., or
- (b) Conducted by contractors other than the supplier ...., or
- (c) Conducted by the utilities' own people ....

2.1.11 IF UTILITY PERSONNEL ARE RESPONSIBLE FOR ALL OR PART OF FUEL INSPECTIONS' WHAT FUNCTIONS ARE THEY ABLE TO PERFORM INDEPENDENT OF SUPPLIERS' PEOPLE OR EQUIPMENT

☐ Video inspection, undemounted bundle	☐ Sipping	☐ Ultrasonic	☐ Profilometry
☐ Surface profilometry, " "	☐ Demounting & visual	☐ Eddy current	☐ Metallography

2.2 TABULAR INFORMATION ON FUEL INSPECTIONS

2.2.1 FOR EACH PLANT SHUTDOWN DURING WHICH FUEL HAS BEEN EITHER REMOVED AND INSPECTED OR SIPPED IN PLACE WITH THE CONCLUSION THAT FAILURES HAD OCCURRED, PLEASE COMPLETE TABLE 2.2.1.

2.2.2 FOR EACH ASSEMBLY IN WHICH FAILURE OF AT LEAST ONE PIN HAS BEEN VISUALLY OR OTHERWISE CONFIRMED, PLEASE COMPLETE A COPY OF TABLE 2.2.2

2.2.3 HAVE ATTEMPTS BEEN MADE TO CORRELATE FAILURE FREQUENCY TO GEOGRAPHIC LOCATION RELATIVE TO CONTROL RODS OR TO RATE OF LOCAL POWER CHANGE, OR OTHER FACTORS? .... ARE THESE STUDIES CONVINCING TO YOU? .... CAN YOU SUPPLY COPIES OF REPORTS? ....

FOOTNOTES

(For use by respondent for additional comments or information)

....

....

....

....

....

3. TECHNICAL INFORMATION FOR SIMULATION OF CORE PERFORMANCE

3.1 GENERAL REACTOR AND CORE DESCRIPTION

3.1.1 Data:

* 1. Plant thermal power rating - MWt ______

2. Primary system piping and instrumentation diagram ___

3. Final safety analysis report, including: ___

   a. The design basis accident descriptions ___

   b. Plant transient analyses ___

4. Technical Specifications - for operation of the reactor, as set forth in the operating license (current version) ___

* 5. Reactor Heat Balance Diagram ___

* 6. Design Data:

   a. Core dimensions - active length ______

   - equivalent diameter ______

   b. Heat transfer area ______

   c. Total weight of UO_2 contained in the core ______

3.1.2 Drawings

3.1.2.1 General core arrangement showing location of:

* (1) Fuel assemblies by region or type, as given in Fig. 1.3.1 ___

* (2) Control rods- indicate control rods which are operated as a bank, as shown in Fig. 1.3.2 ___

* (3) In-core flux detectors ___

(4) In-core temperature detectors ___

(5) Neutron sources ___

(6) Out-of-core flux detectors ___

* (7a) Location of any fixed shim, or burnable poison (PWR) ___

* (7b) Location of any burnable poison curtains (BWR) ___

* (8) Any other non-standard core components which affect the fuel ___

---

*Information required to set up the simulator for the reactor.

3.1.2.2 Reactor Pressure Vessel - dimensioned plan and elevation view of vessel showing as built locations of:

(1) Vessel nozzles ___

(2) In-core chambers and temperature detectors ___

* (3) Active fuel region (elevation) ___

* (4) Control rod location at top and bottom of stroke ___

(5) Out-of-core detectors (elevation) ___

(6) Vessel internals ___

* 3.2 FUEL ASSEMBLY DESCRIPTION

3.2.1 Design Data - List, by fuel type, the as built values for each of the following:

(1) Fuel material and weight per assembly ________

(2) Pellet diameter) ___

(3) Pellet stack height and smear density) as specified in

(4) Clad Material, O.D. and wall thickness) Table 5.2.1 ___

(5) Fuel rod pitch ________

(6) Impurity levels of cladding and fuel in equivalent boron concentrations ________

(7) Weight of heavy metal per assembly by enrichment ________

(8) Spacer weight by material ________

(9) Fuel Assembly weight - total ________

For BWR Only:

(10) Outer dimension of fuel assembly box ________

(11) Fuel assembly box thickness ________

(12) Thickness of narrow water gap ________

(13) Thickness of wide water gap ________

(14) Radius of box corners ________

3.2.2 Dimensioned Drawing for each fuel type showing:

(1) Overal dimensions ___

(2) Active fuel column length and location ___

(3) Tie plate dimensions ___

(4) Spacer locations, material

(5) Flow areas - tie plate, spacer, bare rod ___

(6) Fuel rod diameter and spacing ___

(7) Instrument guide tube locations and dimensions ___

(8) Control rod guide tube locations and dimensions ___

(9) Initial enrichment distribution (as specified in Table 5.2.1) ___

3.2.3 State of fuel from previous cycles - if the simulation does not start from the first operating cycle:

(1) Axial Exposure distribution for each burned assembly ___

(2) Axial Power Distribution for each assembly at end of its last cycle in core ___

(3) Axial Exposure weighted void distribution for each burned assembly (BWR onlv) ___

3.3 REACTIVITY CONTROL SYSTEM

(1) Control rod drive type, notch length and stroke ___

(2) Control rod speed-measured values for scram and normal withdrawal and insertion speeds or times and maximum allowable scram time ___

* (3) Soluble poison concentration as function of exposure ___

* 3.3.1 Control Rod Description, PWR

Dimensioned elevation views showing:

* (1) Shape and configuration ___
* (2) All dimensions necessary to define the location, size and length of all rods in the cluster of the RCC ___
* (3) Poisoned zone dimensions and location relative to unpoisoned portion of the rod ___
* (4) Material composition - structural components and poison ___
* (5) Poison density ___

* 3.3.2 Control rod Description, BWR

* (1) Shape, configuration and axial position relative to active fuel ___
* (2) Cross-section drawing, giving the following dimensions:

Thickness ___

Thickness of control rod outer clad ____________

Total width of control rod blade (half span) ____________

Radius of boron absorber (= inner radius of poison rod) ____________

Outer radius of poison rod pins ____________

Clad thickness of poison rod pins ____________

Pitch of poison rod pins within control rod ____________

Number of poison rods in each control rod blade ____________

Center piece half span ____________

(3) Material composition - structural components and poison:

Absorber material ____________

Density of absorber material, g/cm^3 ____________

Weight fraction of absorber isotope (B^{10}) in absorber material (B_4C) ____________

Clad material tubes ____________

sheets ____________

Density of clad material, g/cm^3 tubes ____________

sheets ____________

* 3.3.3 Burnable poison (BWR)

3.3.3.1 Curtains (if any)

(1) Shape, configuration and axial position relative to active fuel ____________

(2) Cross-section drawing, giving following dimensions and material composition:

Thickness ____________

Width ____________

Effective width of poison area ____________

Absorber material and composition ____________

Density of absorber material ____________

Weight fraction of absorber isotope ____________

Clad material (if present) ____________

Clad density, g/cm^3 ____________

3.3.3.2 Burnable poison in fuel:

Poison material

Poison concentration in fuel rods (wt% Gd_2O_3), as specified in Table 5.2.1 ___

3.3.4 Burnable poison rods (PWR):

(1) Shape, configuration and axial positions relative to active fuel ___

(2) Cross-section drawings giving dimensions and material compositions ___

3.4 THERMAL AND HYDRAULICS DATA

3.4.1 Thermal and hydraulics data, PWR:

(1) Design Data (or actual data if plant is operating)

* a) Reactor pressure (operating and design) - in core outlet plenum ___
* b) Reactor coolant conditions at rated power ___
  * 1) Total core flow rate ___
  * 2) Effective flow rate for heat transfer ___
  * 3) Reactor coolant vessel inlet temperature, ^oC ___
  * 4) Reactor coolant vessel outlet temperature, ^oC ___
  * 5) Core pressure drop at rated power and rated flow ___
  * 6) Assembly pressure drop at rated power and rated flow ___
  * 7) Assembly pressure drop and uncertainty under nominal conditions, at lower and upper nozzle locations, at lower and upper end plate locations, at spacer or grid locations and between spacers ___
  * 8) Void distribution at rated power for highest power fuel assembly ___
  * 9) Effective Doppler temperature (^oC) at rated power ___
  * 10) Volumetric averaged fuel temperature (^oC) as function of power density ___

3.4.2 Thermal and hydraulics data, BWR:

(1) Average fuel temperature as a functions of average power density around nominal power (diagram or table) ________

(2) Hydraulic length per fuel channel, m ________

(3) Pressure in steam dome (upper plenum) ________

(4) Pressure before core inlet (lower plenum) ________

Each type of fuel channel:

(5) One phase inlet pressure drop coefficient $\frac{\Delta p}{\frac{1}{2}\rho V^2}$ ________

(6) One phase lower and upper grid pressure drop coefficient ________

(7) One phase pressure drop coefficient for each spacer ________

At nominal operating conditions:

(8) Fraction of power through cladding, % ________

(9) Fraction of power in bypass flow, % ________

(10) Feedwater temp, ^{o}C ________

(11) Total core flow, kg/sec ________

(12) Bypass flow, % ________

(13) Plenum-plenum core pressure drop, bar ________

(14) Steam quality (weight) at pump inlet, % ________

(15) Void distribution at rated power for highest power fuel channel ________

SECTION 4 - OPERATING HISTORY BY CYCLE

4.1 TOTAL POWER HISTORY

4.1.1 Please supply total power vs. time plot for each operating cycle (see exhibit Figure 4.1.1 for sample). Using information Table 4.1 "Power Shock Severity Sequence" as a guide, number all these "Events" which may have been significant with respect to fuel power shocks during each cycle. Mark each event on the power time plots of exhibit Figure 4.1.1. Prepare a Table of Events (Table 4.1.1).

4.2 SIMULATION OF CORE PERFORMANCE (Core Follow)

4.2.1 For each cycle, provide operating data for a stepwise simulation of core burnup. The actual operating history should be represented by operating periods averaging a core exposure of about 1000 MWD/TU. (For the BWR, preferably corresponding to one rod sequence.) For each step, provide steady state operating data near beginning and end of step (BOS and EOS), which are representative for the operating conditions throughout the step. The information may be supplied in the form of available data sheets from the process computer or other sources giving the following:

a) Time and cycle burnup (MWD/TU).
b) Heat balance data: Thermal power, total core flow rate, feedwater flow and temperature or enthalpy (BWR), core inlet enthalpy and reactor pressure in core outlet plenum (PWR).
c) Control rod positions.
d) Boron concentration (PWR).

4.2.2 At beginning, middle and end of cycle, give a complete set of flux distribution measurements from the traveling fission chamber system and the corresponding operating conditions, in accordance with 4.2.1. [1]

---

[1] These measurements will be used to check the performance of the simulator. (If required, adjustments may be made in the model to correct for discrepancies between calculated and measured power distributions.)

4.3 SIMULATION OF EVENTS

Before and after,"snapshots" of operating data surrounding each important event are required. (Events are identified on the power-time plots.)

4.3.1 For each event, give operating data, as requested under 4.2.1, for steady state operation prior to event.

4.3.2 For each event, provide a record of rod movements and any change in the operating parameters of 4.3.1. Give time when the change took place or if continuous, give time of initiation, estimated rate of change and final value.

4.3.3 Operating data, as requested under 4.2.1, for steady state operation after event.

4.4 POWER DISTRIBUTIONS (if available)

4.4.1 For each total power event, provide: Core maps giving relative assembly power values. (Exhibit 4.2.1.1 and 4.2.1.2 of Section 8 give examples.) Mark each such pair of core maps with the number of the event they depict.

4.4.2 For each local power event involving rod withdrawal or local power change, provide relative assembly power core maps for power distributions before and after the event (per exhibits 4.2.1.1 and 4.2.1.2). In addition, provide before and after relative pin power maps (see exhibit 5.1.1) for typical power distributions in assemblies affected by the change.

4.4.3 On the power-time plots for each cycle which identify events, give a conservative estimate (low side) for the time in which the event took place (minutes). Consider only the time in which total or local power was rising.

4.5 OPERATING RECOMMENDATIONS

4.5.1 As progressive recognition of weaknesses of fuel has occurred, various limitations have been placed upon plant operation aimed usually at minimizing mechanical interaction between fuel and cladding. To allocate fuel failure frequency data properly, it must be interpreted in the context of the operating restrictions in force at the time of failure. In the table below, please indicate which rules were in force for each cycle and, if possible, give the number of the document which identified these rules or criteria.

Cycle No.	*Interim Criteria or Operating Recommendations Governing Rate of Power Change, Control Rod Use, etc.* [1] *(Describe in words the key restrictions)*	*Reference Document Number, if any*
1	______________________	__________
	______________________	__________
	______________________	__________
2	______________________	__________
	______________________	__________
	______________________	__________

4.6 FISSION GAS RELEASE HISTORY

4.6.1 Total I_{131} activity in primary system or off gas vs. time on plots for each cycle, include as Figure 4.4.1 in reply.

4.6.2 Calculated I_{131} release rate vs. time on plots for each cycle, include as Fig. 4.4.2 in reply.

4.6.3 Calculated number of leaky pins vs. time as Figure 4.4.3 in reply. Explain model used for this calculation as attachment to the Figure.

(1) These supplier originated guides do not legally restrict operation or affect guarantees. They are recommended to gain extra margin, etc., etc. The key question is: Did this particular cycle stay within such a guideline, and if so, which one?

____

____

TABLE 4.1.1.__ TABLE OF EVENTS REFERRED TO POWER TIME PLOTS OF FIGURES 4.1.1.__

CYCLE NO. : ________

FROM (Date): ________

TO (Date): ________

Event No.	*Date of Occurrences*	*Nature of Event*	*Severity of Event* (1)	*Activity Release Rates*	
				Before	*After*

FOOTNOTES:

(1) From Table 4.5 (Power Shock Severity Sequence)

INFORMATION TABLE 4.1. POWER SHOCK SEVERITY SEQUENCE FROM 1 (WORST) TO 36 (LEAST)

Time at Low Power Prior to Power Increase

Time in which Power Increase Took Place	1 DAY	1 WEEK	1 MONTH	3 MONTHS	% Power Change Band from: x to: 100% Rated
5 MIN.	28	15	14	13	85 - 100
5 MIN.	31	22	5	4	70 - 100
5 MIN.	34	25	10	1	50 - 100
1 HR.	29	18	17	16	85 - 100
1 HR.	32	23	7	6	70 - 100
1 HR.	35	26	11	2	50 - 100
1 DAY	30	21	20	19	85 - 100
1 DAY	33	24	9	8	70 - 100
1 DAY	36	27	12	3	50 - 100

EXPLANATIONS:

An "event" with some probability of producing power shock failures has occurred each time the LHGR (local heat generation rate, W/cm or kW/ft.) everywhere in the core is raised to a level not reached previously (or not reached for some time previously). The failure-importance of an event is determined by the amount of fuel involved (cm or ft.) and the magnitude of the power shock it has experienced. Power shock ($\Delta Q / \Delta Q_{failure}$) is a complex function of the following variables: The magnitude of local power increase (a statistical distribution); the time in which the power increase took place; and the prehistory of the fuel involved (time at, and level of, previous power).

An operating cycle is impossible to characterize perfectly with respect to power shock, without microscopic analysis of every change involving total power, rod motion, fuel shuffling, and power redistribution during Xenon transients. It is possible, however, at lower precision to account for major shocks received generally or locally by the fuel and, thereby, predict 80 - 90% of failures which should have occurred during the cycle under consideration. The approach to this problem taken herein is to identify and characterize "important" events which can be characterized by the time in which they occurred and the power distributions existing before and after the events. All of the following events are "important" and should be marked on the power-time plots of Figure 4.1.1.x.:

- Any return to full power after fuel has been rezoned or shuffled into regions of higher relative power.
- Any return to full power after control blade (BWR) patterns have been changed (e.g., after rod sequencing in BWR).

There are two types of events whose importance can be appraised by reference to the table below: (1) Any

Table 4.1 (cont.)

increase in local power (e.g., caused by rod withdrawal) preceded by a period at lower local power. The table gives the relative severity of 36 shock situations without reference to the total length of fuel affected. In the most severe 18 events depicted, 93% of all failures have been produced. The respondent case gauge the importance of a particular rod withdrawal by referring to the table. For example: the local powers in the most affected fuel pin(s) double when Bank D rods are withdrawn from 180 to 200 units (i.e., % power change band 50 - 100). The withdrawal took 1 hour, and the rods had been in position 180 for a week previously. Reference to the table shows (Col. 2, Row 6) that this event has severity position 26 and is not very important. If the rods had been in for a month, however,(Col. 3, Row 6), this would have been an important event (see No. 11).

In analysing a cycle, the respondent is asked to decide to which level he will report events. Reporting events less severe than the 18th in the table above is unnecessary for this tudy. However, reporting only the more severe events is none the less useful (if detailed analysis of events below the 6th or 7th severity level is considered excessively time consuming by the respondent, an appropriate correction factor will be applied to the cycle failure predictions).

SECTION 5 - MANUFACTURING AND DESIGN DATA FOR FUEL TYPES EMPLOYED

5.1 IDENTIFICATION OF FUEL ASSEMBLIES BY TYPE, ENRICHMENT SPLIT, ROD DESIGN SPLIT, POWER DISTRIBUTION

5.1.1 FILL IN A COMPLETE FIGURE 5.1.1 FOR EACH TYPE AND BATCH LISTED IN TABLE 1.2.1

(a) In diagram (a) identify rods that are identical with respect to enrichment, dimension and burnable poison by letter designation (For sample, see Exhibit 5.1.1).

(b) In diagram (b) give design relative powers of rods with associated control rods out (at zero burnup).

(c) In diagram (c) give design relative powers of rods with associated control rods in (at zero burnup).

(d) In diagram (d) give design relative powers with associated control rods out (at high burnup).

(e) Provide any additional data describing relative power shifts during cycle.

5.2 DESIGN OF ASSEMBLIES AND RODS

FOR EACH TYPE AND BATCH LISTED IN TABLE 1.2.1 FILL OUT A COPY OF TABLE 5.2.1. THE COLUMN HEADINGS ARE IDENTICAL TO IDENTIFICATIONS OF IDENTICAL ROD GROUPS USED IN FIGURES 5.1.1.X.

5.3 PROPERTIES OF MATERIALS

5.3.1 IRRADIATED PROPERTIES OF ZIRCALOY EMPLOYED

The local ductility of irradiated Zircaloy tubing is one of the most critical of the factors determining wheter PCI failure will occur. Experts disagree on the type of test that is most meaningful (tensile, burst, biaxial, etc.) as a measure of the PCI resistance of local cladding. The stresses involved are complex. Both biaxial and burst test data appear to be more representative than simple tensile data. Even more important than the average value of strain to failure is the statistical variation in that value. Please attach as Figures 5.3.1.1 to 5.3.1.X irradiated stress strain data for can materials and treatments employed for the fuel types that have been used.

SECTION 6 - FIGURES

This section is reserved for insertion of Figures included in the completed questionnaire by the respondent. Samples of type of Figures desired are included under the same numbers in Section 8, "Exhibits".

1.2.1.1 to 1.2.1.X	Assembly Layout Drawings per Sample of Exhibit 1.2.1.
1.3.1.1 to 1.3.1.X	Core Maps Showing Location of Fuel by Cycle per Sample of Exhibit 1.3.1.
1.3.2	Core Maps Identifying Control Rod Arrangement.
3.1.1.1 to 3.1.1.X	Power vs. Time Plots
3.2.1.1 to 3.2.1.X	Relative Power Core Maps
3.2.2.1 to 3.2.2.X	Axial Power Distribution
3.3.1.1 to 3.3.1.X	Rod Position Histories
3.4.1.1 to 3.4.1.X	Iodine 131 Activity vs. Time
3.4.2.1 to 3.4.2.X	Iodine 131 Release Rates
3.4.3.1 to 3.4.3.X	No. of Leaks vs. Time from I_{131}
5.1.1.1 to 5.1.1.X	Identification of Rod and Power Splits (Blanks Included)

FIGURE 5.1.1 - PICTORIAL DEFINITION OF FUEL ROD SPLIT AND RELATIVE POWER DISTRIBUTION

(One required for each fuel type and batch shown in Table 1.2.1)

TYPE DESIGNATION: ....

SERIAL NO'S: ....

BATCH: ....

AVERAGE ENRICHMENT ....%

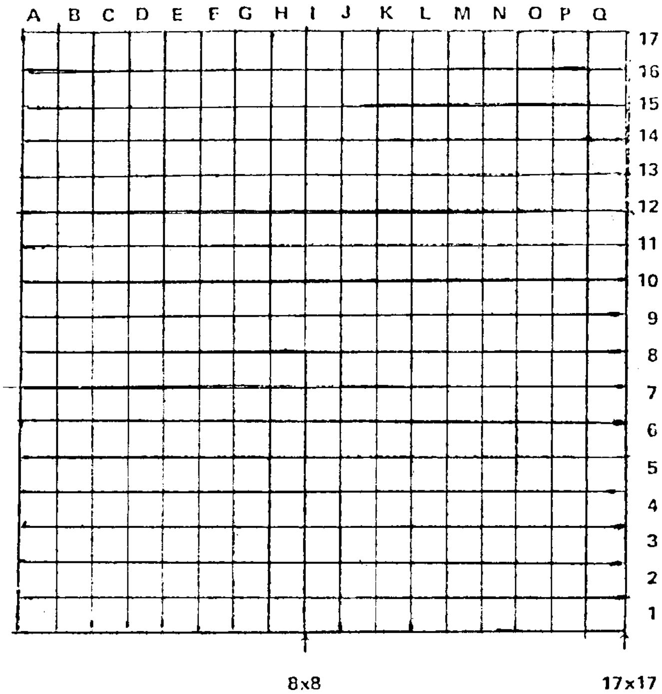


Fig. 5.1.1.a - Identification of Groups of Rods that are identical in Dimension, Enrichment, and Burnable Poison Content (see Exhibit 5.1.1 for Sample) (1)

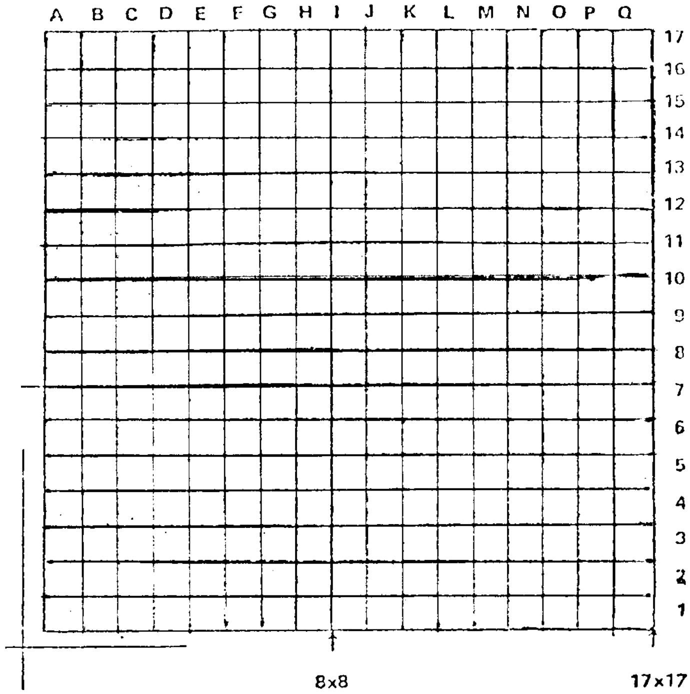


Fig. 5.1.1.b - Give Design Relative Powers of Pins at Cycle Start with Control Rods Fully Out (see Exhibit 5.1.1 for Sample)

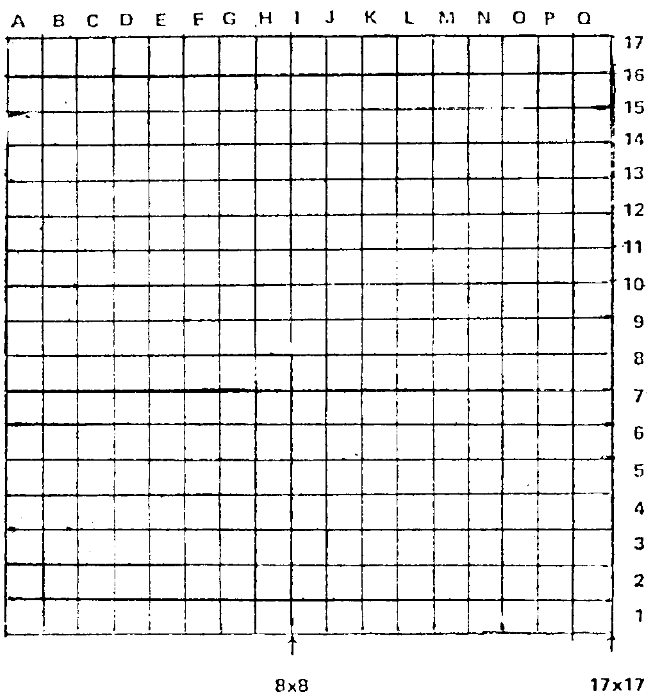


Fig. 5.1.1.c - Design Relative Powers at Cycle Start. Associated Control Rods Fully In.

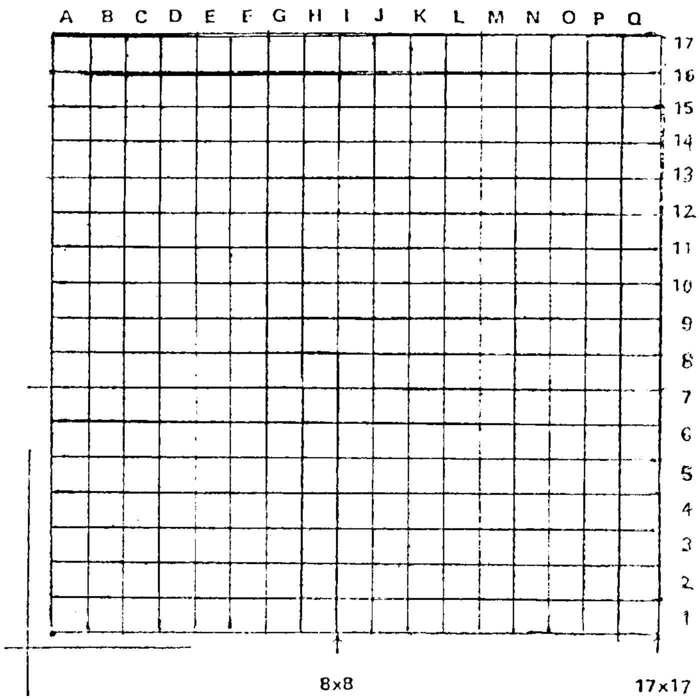


Fig. 5.1.1.d - Design Relative Powers Late in Irradiation - Control Rods Out. Burnup .... MWd/kg

(1) Occasionally enrichment is split axially. If differences exist, please show as separated rod types and define in Table 5.2.1.

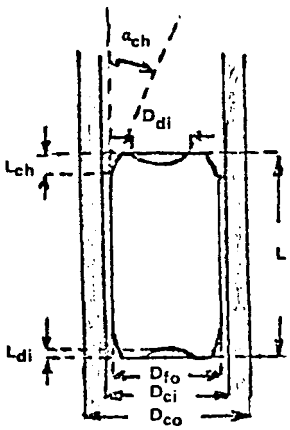
TABLE 2.2.1 TABULAR SUMMARY OF INSPECTION RESULTS FOR FUEL DISCHARGED AT END OF CYCLE

Date of Discharge Operation: From: ______ ; ______ Sheet ___ ___

List Type and Serial Numbers of Assemblies Believed to Contain at least 1 Failed Pin		I IN-CORE SIPPING BEFORE REMOVAL		II BASIN SIPPING AFTER DISCHARGE		III VISUAL EXAMINATION BEFORE DEMOUNTING			IV INSPECTION AFTER DEMOUNTING							V GIVE ESTIMATED NUMBER OF FAILED PINS IN THIS ASSEMBLY (1)	VI IS FURTHER INSPECTION SCHEDULED ON THIS ASSY? (2)	VII WHAT IS CURRENTLY BELIEVED CAUSE OF THE FAILURES? (3)	VIII FINAL DISPO-SITION OF ASSY	IX WHICH OUT-SIDE FIRM(S) DOING INSP. WORK?
		A	B	A	B	A	B	C	A	B	C What Type Inspections									
TYPE	SER.No.	Was it Done?	Did it show Failure?	Was it Done	Did it show Failure	Was it Done?	Do TV Tapes Exist?	Did any Outside Rods have Holes?	Was it Done?	Pertinent Report Covering Insp.	Eddy Current	Ultra-sonic	Visual	Hot Lab.	Pro-file					

(1) If not acutally known, indicate basis for estimate in another footnote below
(2) Interaction, Densification. Hydriding, Man. Defect, etc.
(3) Has assembly been (a) rebuilt & reinserted? (b) Removed permanently? (c) Held in Storage?

Table 5.2.1. ___ DESIGN DETAILS BY ASSEMBLY TYPE



NOTES:

(1) Complete this table for each fuel type and batch listed in Table 1.2.1

(2) The columns refer to groups of rods tabulated as identical in Figures 5.1.1.x(a), see rod split figure.

(3) Where an entry is identical to one on its left, left insert ditto mark: " „ "

KEY:

(1) Assembly Type Designation ________ (Table 1.2.1)

(2) Batch Identity ________ (Table 1.2.1)

(3) Average Enrichment ________

(4) Serial Numbers Covered ________

(5) Which Figures 5.1.1.x corresponds ________ (Value of x)

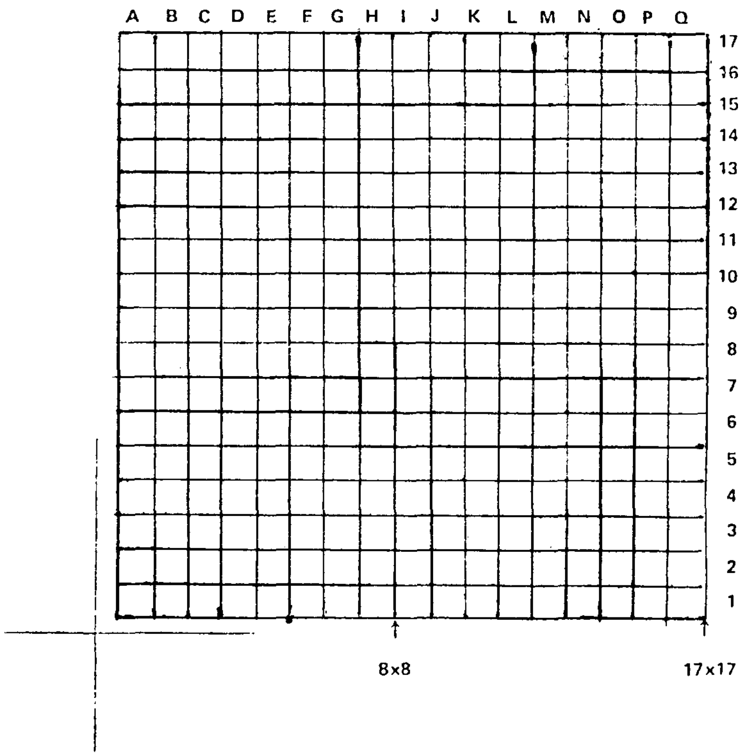
PARAMETER	FUEL ROD GROUP IDENTIFICATION FROM FIG. 5.1.1									
	A	B	C	D	E	F	G	H	I	J
Pellet Length										
Can, O.D.										
Can I.D.										
Pellet O.D.										
Diametral Gap										
Gap Tolerance, % of Gap										
Dish Diameter										
Dish Diametral Fraction										
Dish Depth										
Are both ends dished?										
Chamfer Angle										
Chamfer Length, Axial										
Fuel Density Immersion										
Fuel Density by Geometry & wt.										
Density Tolerance, % of Num.										
Fraction of True Cylinder Removed by Dish										
Fraction of True Cylinder Removed by Chamfer										
Sintering Temperature										
Sintering Time										
Enrichment										
Weight% Gadolinium										
Cladding Material, Type										
Cladding Material, Spec. No.										
Is Cladding Pickled?										
Is Inside Sandblasted?										
Is Cladding Autoclaved?										
Largest Acceptable Defect										
Filler Gas Pressure										
Free Gas Volume in Rod										
Water Limit in Pellets										
Water Limit in Whole Rod										
Active Fuel Length										
Is Enrichment Split Axially? (1)										

(1) If axial enrichment or poison split exists, give details in footnote to section 5.

TABLE 2.2.2 FAILURE DATA FOR ASSEMBLY

TYPE:__________ SERIAL NO. ______

Sheet_____ of_____



I. GENERAL

A. Has the assembly been disassembled? __________

B. Or is the examination done on undemounted assembly? __________

C. Is examination completed, or what else will be done? __________

D. Are the failed rods listed below the total number of failed rods in the assembly? __________

E. If not, give best estimate of total number of failed rods. __________

II. DETAILED RESULTS FOR RODS INSPECTED

ROD POSITION	AXIAL FAILURE POSITION*	TYPE OF EXAMINATION PERFORMED: VISUAL	EDDY CURRENT	DESTRUCTIVE P.I.E.	DIAMETER PROFILOMETRY	AXIAL AVG. BURN-UP OF ROD MWD/KG	APPEARANCE OF FAILURE BLISTER, PCI CRACK, COLLAP ETC.
sample B-3	0.4 to 0.6	✓	✓	Scheduled	.0025"	30.2	Bamboo ridges, PCI crac

III. SUPPLEMENTARY DATA

A. Supply or identify pertinent reports covering the inspection: __________

B. Give power history, (average) along rod and axial distribution for each of the failed rods or the total assembly (use separate sheet).

C. Give position history of associated control rod(s) (use separate sheet).

* As fraction of actual fuel length from bottom and up

* Give largest diametral ridge height ($D_{max} - D_{avg}$)

SECTION 8 - EXHIBITS

In this section examples are provided for figures of the type requested in writing in the questionnaire. The samples are an aid in definition of the material requested and which could equally well be transmitted in entirely different format.

EXHIBIT 1.2.1 TYPICAL ASSEMBLY DRAWING

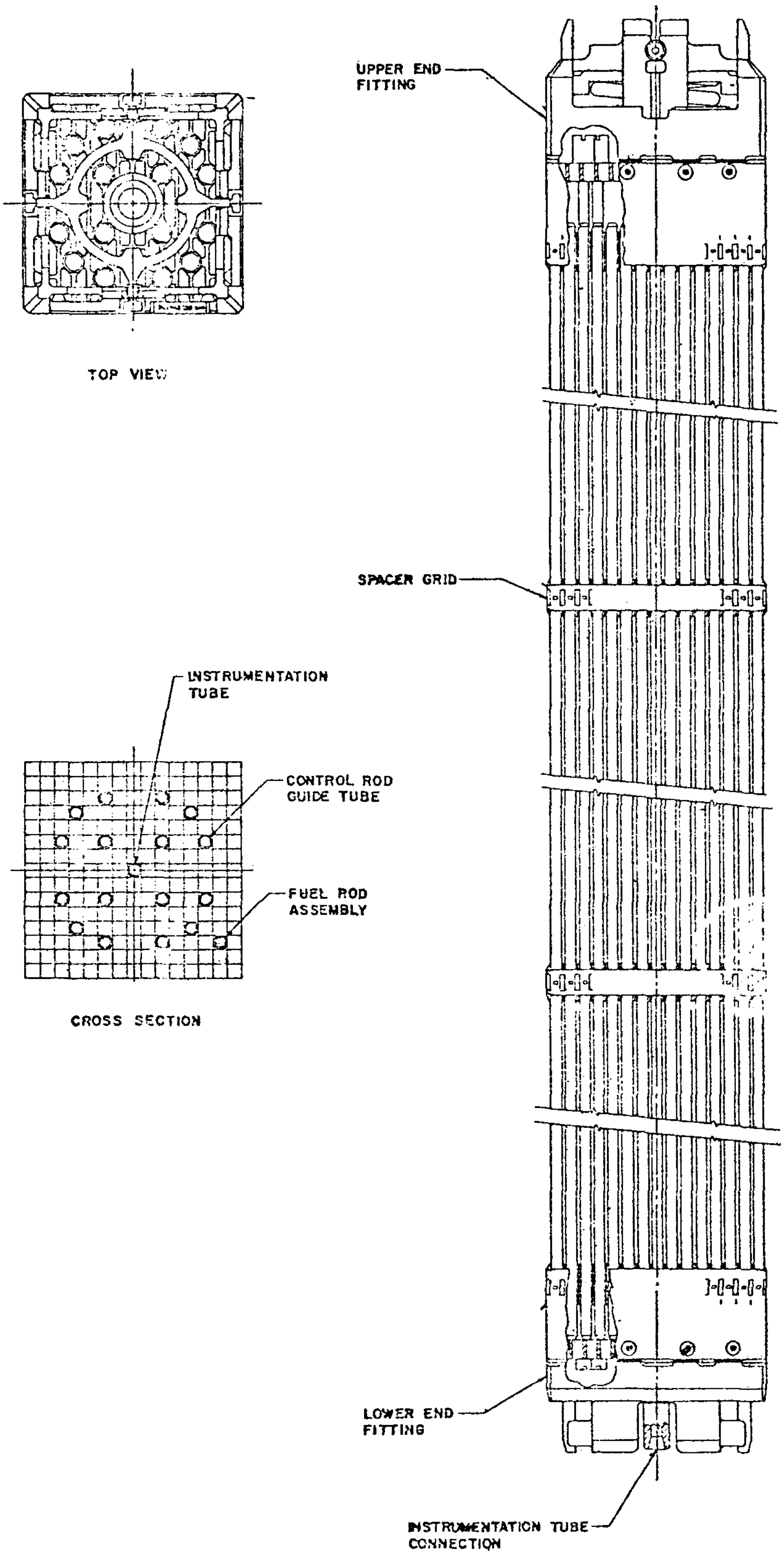
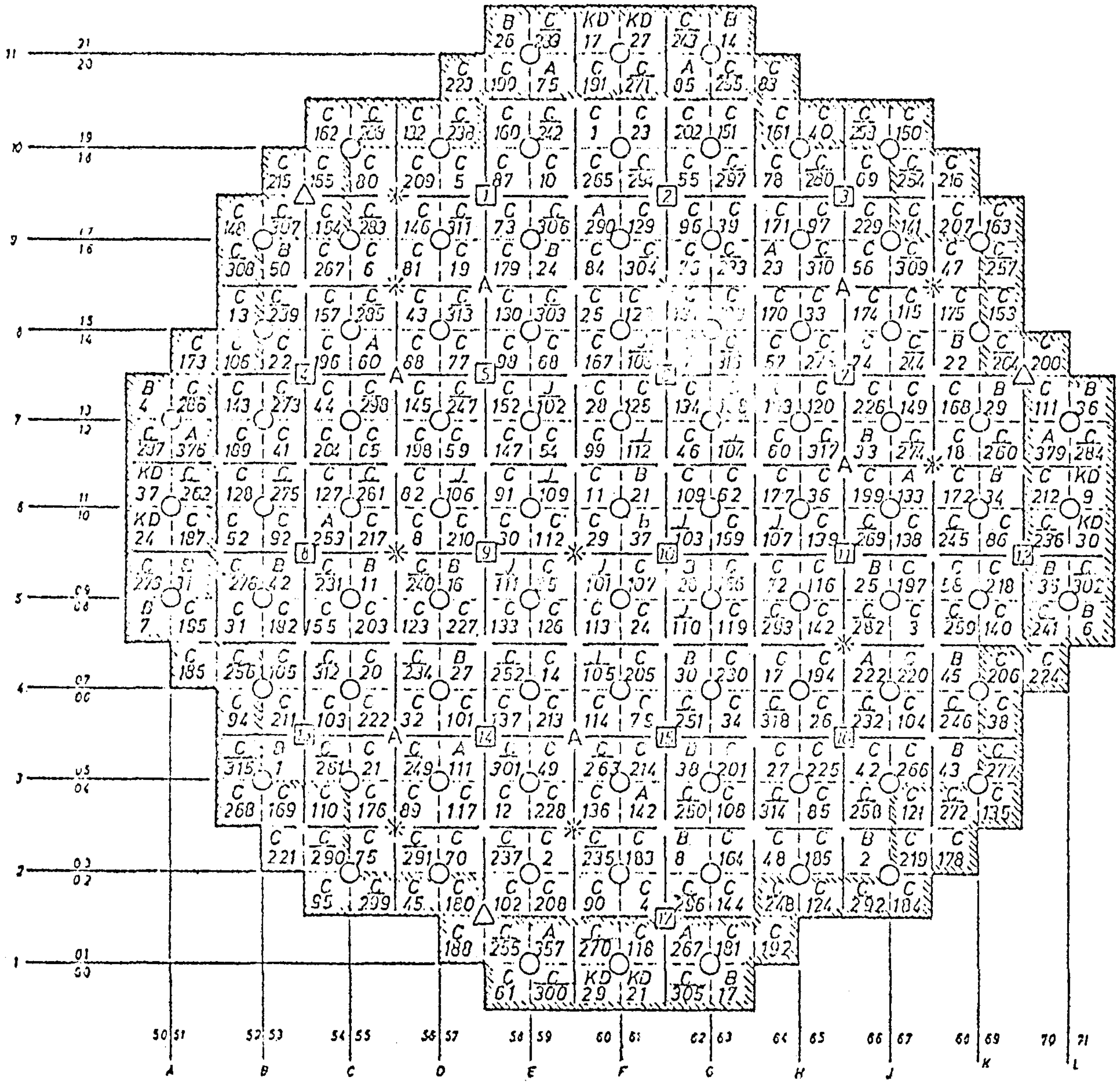


Exhibit 1.3.1. - Fuel Location Core Map



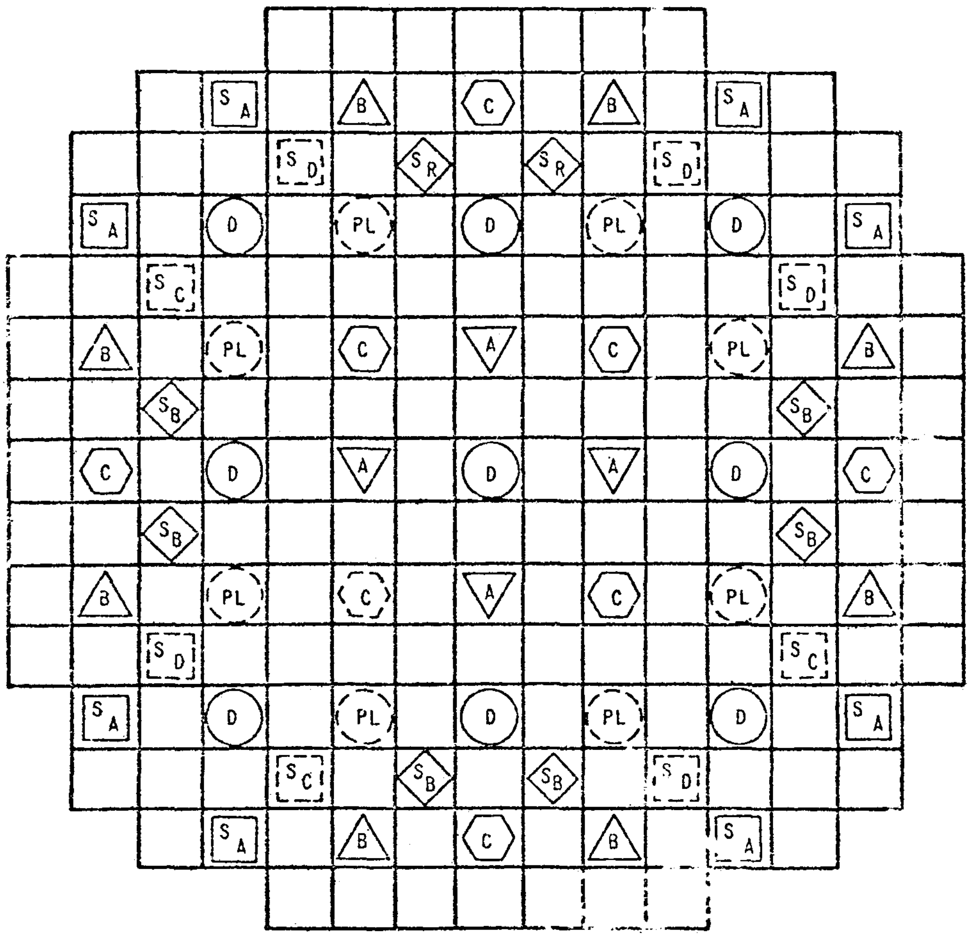
Reactor ______

Cycle ______

Date Start ______

Date End ______

Exhibit 1.3.2.1- Control Rod Configuration - PWR



	FUNCTION	NUMBER OF ROD CLUSTERS
SHUTDOWN BANK	S_A	8
SHUTDOWN BANK	S_B	8
SHUTDOWN BANK	S_C & S_D	4 & 4
CONTROL BANK	A	4
CONTROL BANK	B	8
CONTROL BANK	C	8
CONTROL BANK	D	9
PART-LENGTH	PL	8

Exhibit 3.1.i - Total load vs time, cycle 3 Start: 1/4/69

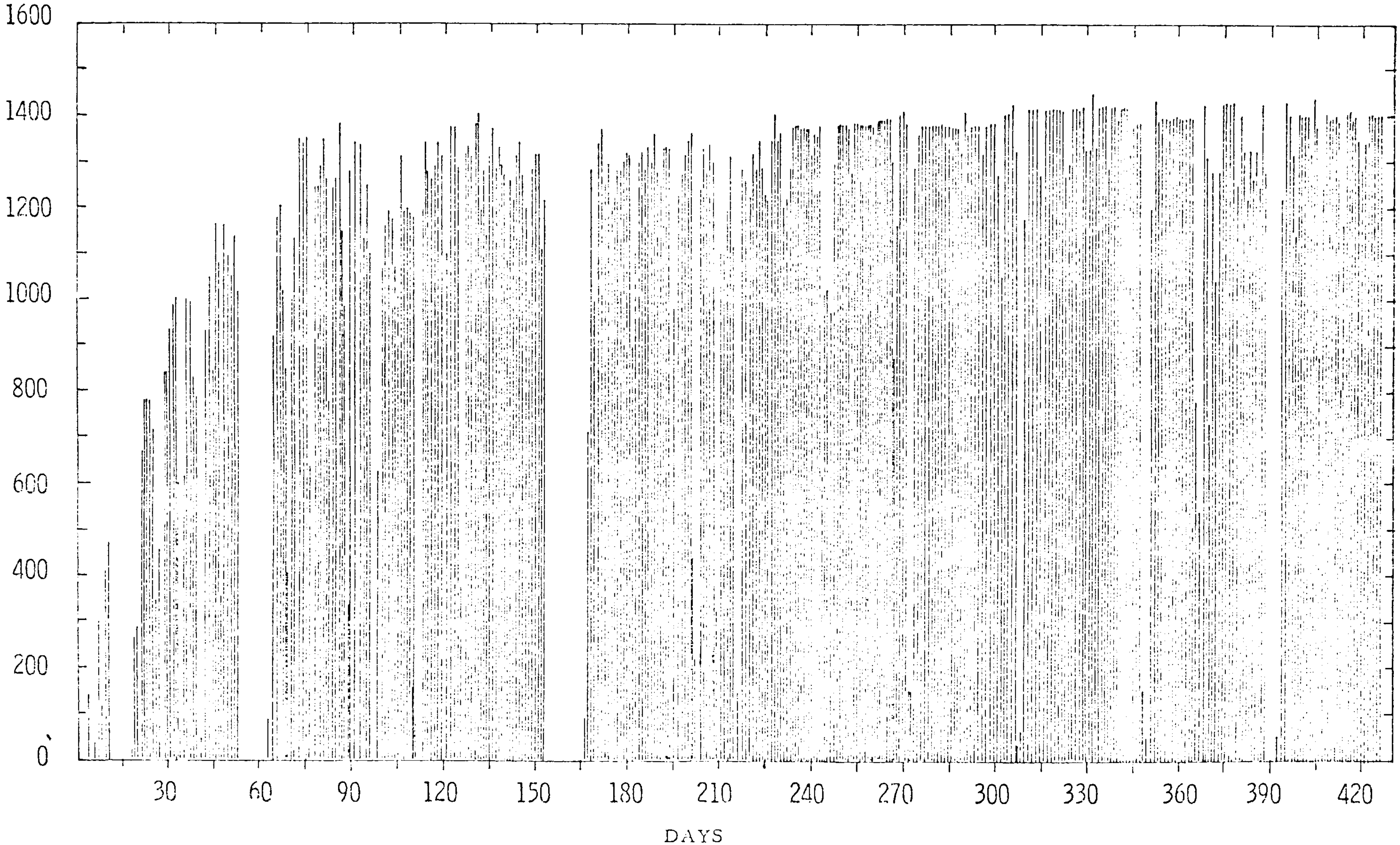


Exhibit 3.2.1.1 Sample of Relative Power Distribution Map

Date: ______________

Cycle No.: ______________

Power Level: ______________

Rod Position Summary (1)

____

____

____

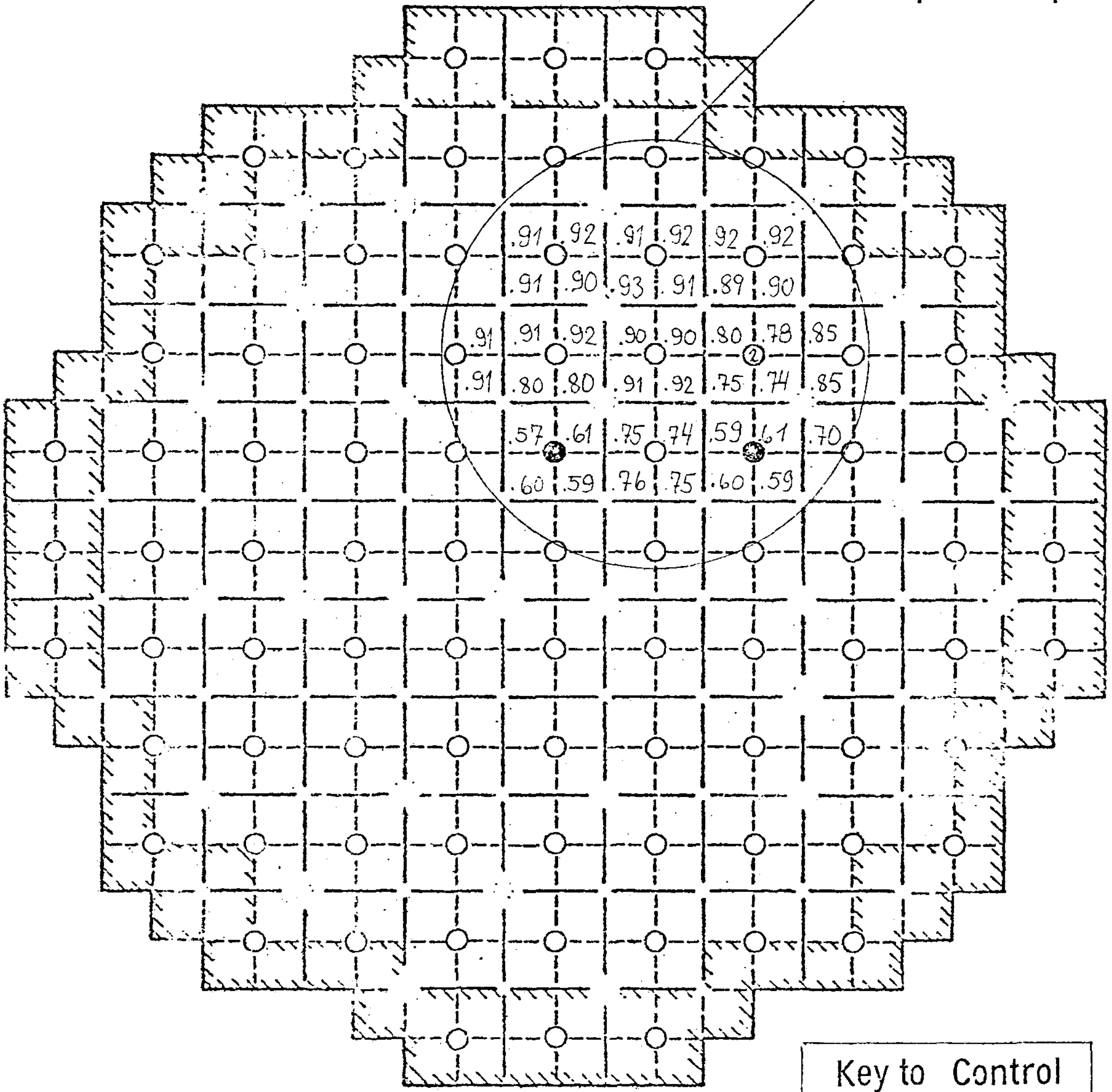
Time in Cycle: ______________

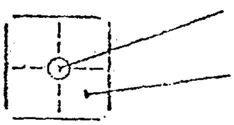
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							.59	.92	.59						
2					.64	.93	.96	.89	.96	.93	.64				
3				.70	.98	1.03	1.13	1.11	1.13	1.03	.98	.70			
4			.70	.98	1.05	1.18	1.13	1.05	1.13	1.18	1.05	.95	.70		
5		.64	.98	1.05	1.18	1.14	1.19	1.07	1.19	1.14	1.18	1.05	.98	.64	
6		.93	1.03	1.18	1.14	1.06	1.13	1.19	1.13	1.06	1.14	1.18	1.03	.93	
7	.59	.96	1.13	1.13	1.19	1.13	1.17	1.05	1.17	1.13	1.19	1.13	1.13	.96	.59
8	.92	.89	1.11	1.05	1.07	1.19	1.05	1.00	1.05	1.19	1.07	1.05	1.11	.89	.92
9	.59	.96	1.13	1.13	1.19	1.13	1.17	1.05	1.17	1.13	1.19	1.13	1.13	.96	.59
10		.93	1.03	1.18	1.14	1.06	1.13	1.19	1.13	1.06	1.14	1.18	1.03	.93	
11		.64	.98	1.05	1.18	1.14	1.19	1.07	1.19	1.14	1.18	1.05	.98	.64	
12			.70	.98	1.05	1.18	1.13	1.05	1.13	1.18	1.05	.95	.70		
13				.70	.98	1.03	1.13	1.11	1.13	1.03	.98	.70			
14					.64	.93	.96	.89	.96	.93	.64				
15							.59	.92	.59						

(1) Refer for Rod Configuration map given under question 1.3.2

Exhibit 3.2.1.2 Relative Power and Control Rod Position BWR Example

Sample of Completed Map





Key to Control Rod Position

○ = out
● = in
⑧ = 80% in
② = 20% in
etc.

Exhibit 3.2.2.-Sample Axial Flux Distribution