GEORGIA INSTITUTE OF TECHN Y OFFIC CONTRACT ADMINISTRATION

PROJECT ADMINISTRATION DATA SHEET

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[x] ORIGINAL [ ] REVISION NO. ____

Project No. E-25-627 (formerly E-25-692) GTRI/GTEX DATE 8/8/84

Project Director: Dr. John M. Kallfelz School/Lab: ME/NE&HP

Sponsor: Martin Marietta Energy Systems, Inc. (formerly Union Carbide Corp., Nuclear Division); Oak Ridge, TN 37830

Type Agreement: Supplement No. 9 to Proj. Auth. #X01 under B.A. #7802

Award Period: From 7/1/84 To ~~6/30/85~~ 12-31-85 (Performance) 7/30/85 (Reports)

Sponsor Amount:	This Change	Total to Date
Estimated:	$ 31,685	$ 31,685
Funded:	$ 31,685	$ 31,685

Cost Sharing Amount: $ n/a Cost Sharing No: ____

Title: "Analytic Reactor Physics and Design Investigations"

ADMINISTRATIVE DATA OCA Contact Lynn Boyd x4820

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Engineering Physics Division
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37830

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Oak Ridge, TN 37830

Defense Priority Rating: n/a Military Security Classification: n/a

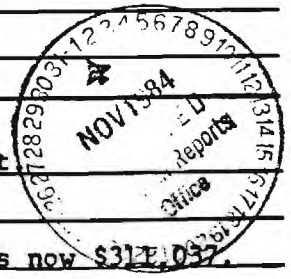
(or) Company/Industrial Proprietary: n/a

RESTRICTIONS

See Attached n/a Supplemental Information Sheet for Additional Requirements.

Travel: Foreign travel must have prior approval – Contact OCA in each case. Domestic travel requires sponsor approval where total will exceed greater of $500 or 125% of approved proposal budget category.

Equipment: Title vests with ____



COMMENTS:

Supplemental Agmt. #9 adds $31,685 for additional year of effort.

Total amount of subcontract (including previous project no.s) is now $317,037.

This project follows E-26-B04.

Sponsor I.D. #02.240.001.84.003

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FORM OCA 4:383

EORGIA INSTITUTE OF TECHNOLOGY OFFICE OF CONTRACT ADMINISTRATION

SPONSORED PROJECT TERMINATION/CLOSEOUT SHEET

Date 8/11/87

ject No. E-25-627 School/~~Lab~~ ME

ludes Subproject No.(s) N/A

ject Director(s) J. Kallfelz GTRC / ~~GIT~~

onsor Martin Marietta Energy System, Inc. (formerly Union Carbide Corp., Nuclear Division); Oak Ridge, TN 37830

le " Analytic Reactor Physics and Design Investigations"

ective Completion Date: 12/31/85 (Performance) 1/31/86 (Reports)

nt/Contract Closeout Actions Remaining:

- [ ] None
- [X] Final Invoice or Final Fiscal Report
- [X] Closing Documents
- [X] Final Report of Inventions
- [X] Govt. Property Inventory & Related Certificate
- [ ] Classified Material Certificate
- [ ] Other

ntinues Project No. E-25-692 Continued by Project No.

PIES TO:

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E-25-627

Georgia Institute of Technology

A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA

SCHOOL OF MECHANICAL ENGINEERING

October 23, 1984

Please reply to:

NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

MEMORANDUM

TO: D.G. Cacuci, ORNL

FROM: J.M. Kallfelz, L.A. Belblidia and J.N. Davidson

SUBJECT: Progress Report for ORNL Subcontract 7802 Month of September 1984

Accomplishments During Report Period

1. J.M. Kallfelz, L.A. Belblidia and J.N. Davidson attended an EPRI - sponsored seminar on modular system techniques. This seminar, held September 26-28, covered techniques for modeling the dynamic performance of nuclear and fossil-fired power plants. Considerable information was disseminated concerning code development and validation, model standardizations, simulation language and utility applications, all of interest in our development of modules for the DSNP[1-3] code for PWR transient analysis.

2. Together with other Georgia Tech staff members, a proposal was written to the NSF. One of the technical areas in this proposal involves transient power plant simulation with the DSNP code, which would involve a cooperative effort with ORNL.

3. Work continued on calculating a loss of offsite power (LOOP) ATWS case[4] using the version of RELAP-3B[5] running at Georgia Tech. The results of this run will be used to compare with DSNP calculations for the same case.

REFERENCES

1. D. Saphier, "A Special Purpose Simulation Language for Nuclear Power Plants", in Simulation of Systems '79, L. Dekker et al., Eds., North-Holland Publishing Co., p. 1055, 1980

2. D. Saphier and J.T. Madell, "The DSNP Simulation Language and Its Application to LMFBR Transient Analysis", Nucl. Tech., 56, p. 493, March 1982

3. D. Saphier, "The Simulation Language of DSNP", ANL-CT-77-20, Rev 02, Argonne National Laboratory, September 1978

4. "Selected ATWS Calculations for three PWR Designs", Battelle Columbus Report, December 1982

5. BNL-NUREG-22011, "User's Manual for RELAP3B-MOD 110, "A Reactor System Transient Code", Department of Applied Science, Brookhaven National Laboratory

E-25-627

Georgia Institute of Technology

A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA

SCHOOL OF MECHANICAL ENGINEERING

Please reply to:

NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

January 24, 1985

MEMORANDUM

TO: D.G. Cacuci, ORNL

FROM: J.M. Kallfelz, L.A. Belblidia and J.King

SUBJECT: Progress Report for ORNL Subcontract 7802, Months of October, November and December 1984.

Accomplishments during Report Period

Efforts continued on debugging the RELAP - 3B code[1] and a PWR model for calculating reactor transients. Attached are some results for a Loss-of-Feedwater (LOFW) transient. Such results will be used to test PWR modules being developed at Ga. Tech for the DSNP simulation language code.[2-4]

Project funding for this task has been almost completely expended, but this work will be continued at Ga. Tech. with internal support.

Telephone: 404-894-3720 Telex: 542507 GTRIOCAATL Fax: 404-894-3120 (Verify: 404-894-4850)

REFERENCES

1. BNL-NUREG-22011, "User's Manual for RELAP3B-MOD 110, "A Reactor System Transient Code", Department of Applied Science, Brookhaven National Laboratory

2. D. Saphier, "A Special Purpose Simulation Language for Nuclear Power Plants", in Simulation of Systems '79, L. Dekker et al., Eds., North Holland Publishing Co., p. 1055, 1980

3. D. Saphier and J.T. Madell, "The DSNP Simulation Language and Its Application to LMFBR Transient Analysis", Nucl. Tech., 56, p. 493, March 1982

4. D. Saphier. "The Simulation Language of DSNP", ANL-CT-77-20, Rev 02, Argonne National Laboratory, September 1978

Loss of Feedwater Transient

Sequence of Events

0–10	Sec	Steady state operation at 3411 MWth
10–14	Sec	Main feedwater ramps to zero Steam generator secondary water level starts dropping
30–50	Sec	Pressure in primary rising Coolant density in core dropping Coolant level in pressurizer rising Power dropping due to negative void coefficient
45	Sec	2 pressurizer PORVs open Coolant density and power drop faster
75	Sec	Pressurizer "goes solid" with liquid water; problem terminated

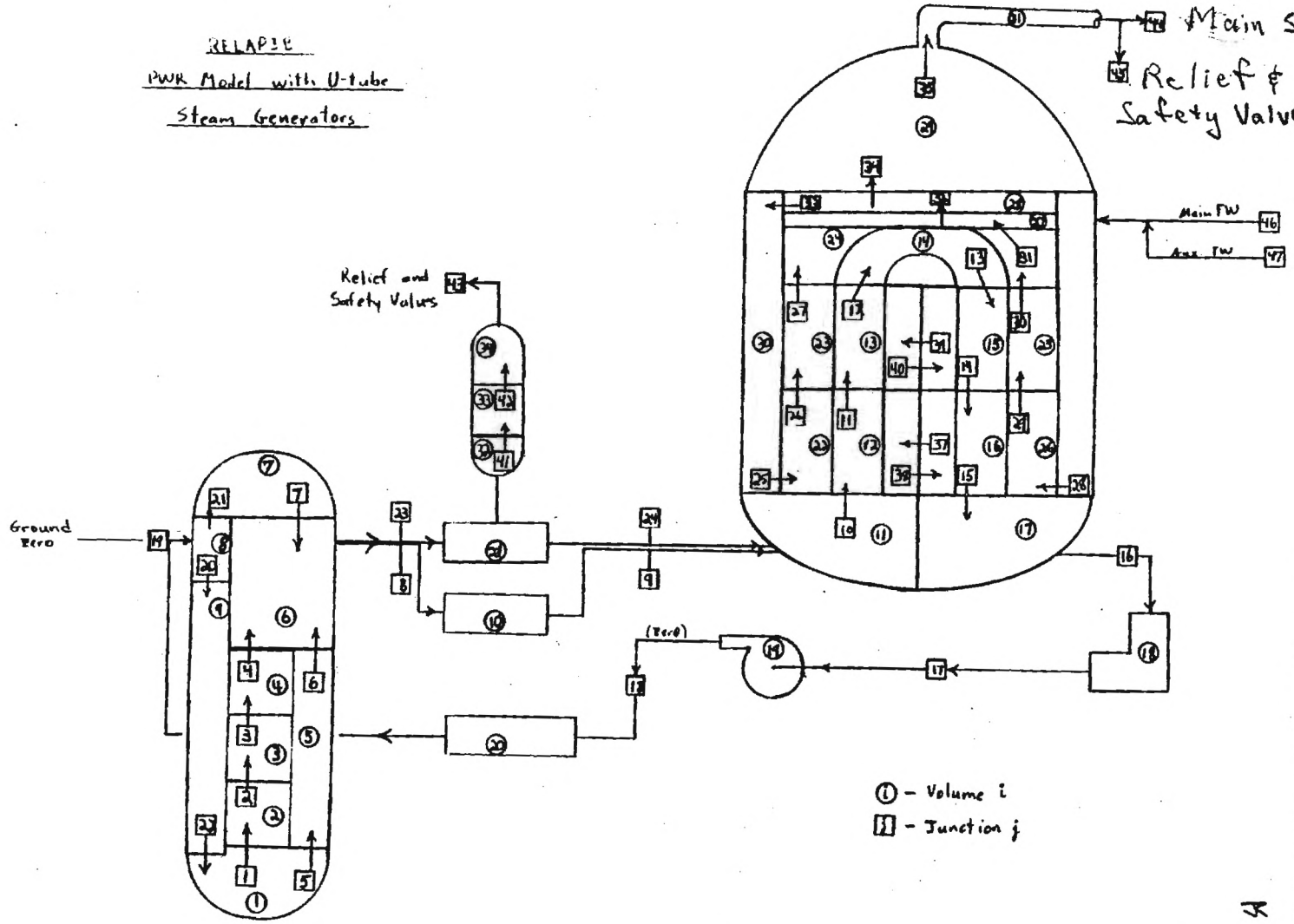

Fig. 1 RELAP 3B MODEL OF PWR - FOR LOFW TRANSIENTS

FIG 2
RELAP 5B
SN SG PWR
LOFW-1

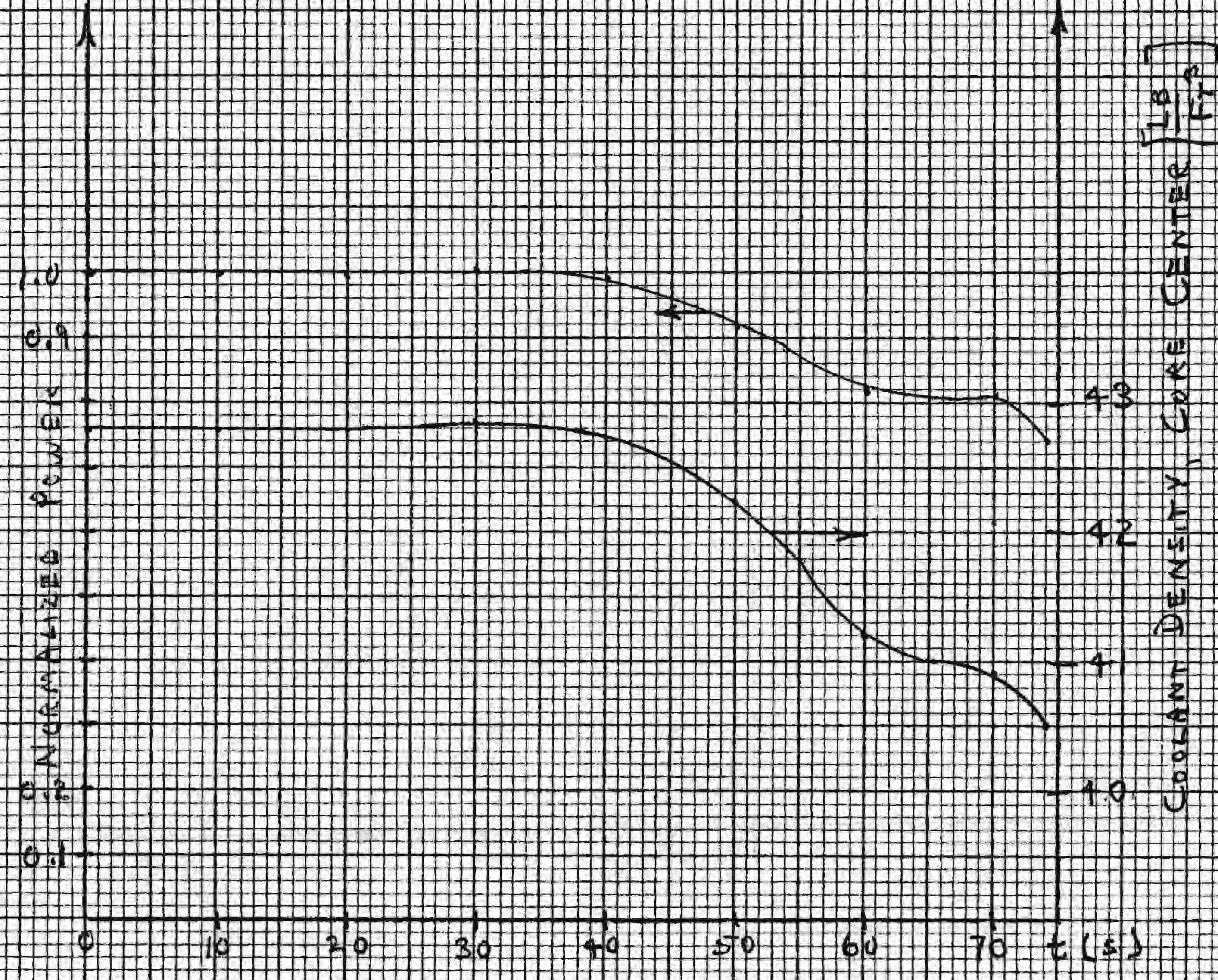

FIG 3

RELAPSB

SNSG-PWR

LOFW-1

FIG 4
RELAP 3B
5N 56 PWR
LOEW-1
VOID & DOPPLER REACTIVITY [$]
0.4
0.3
0.2
0.1
0
-0.1
-0.2
-0.3
-0.4
DOPPLER
VOID
10
20
30
40
50
60
70
t (s)

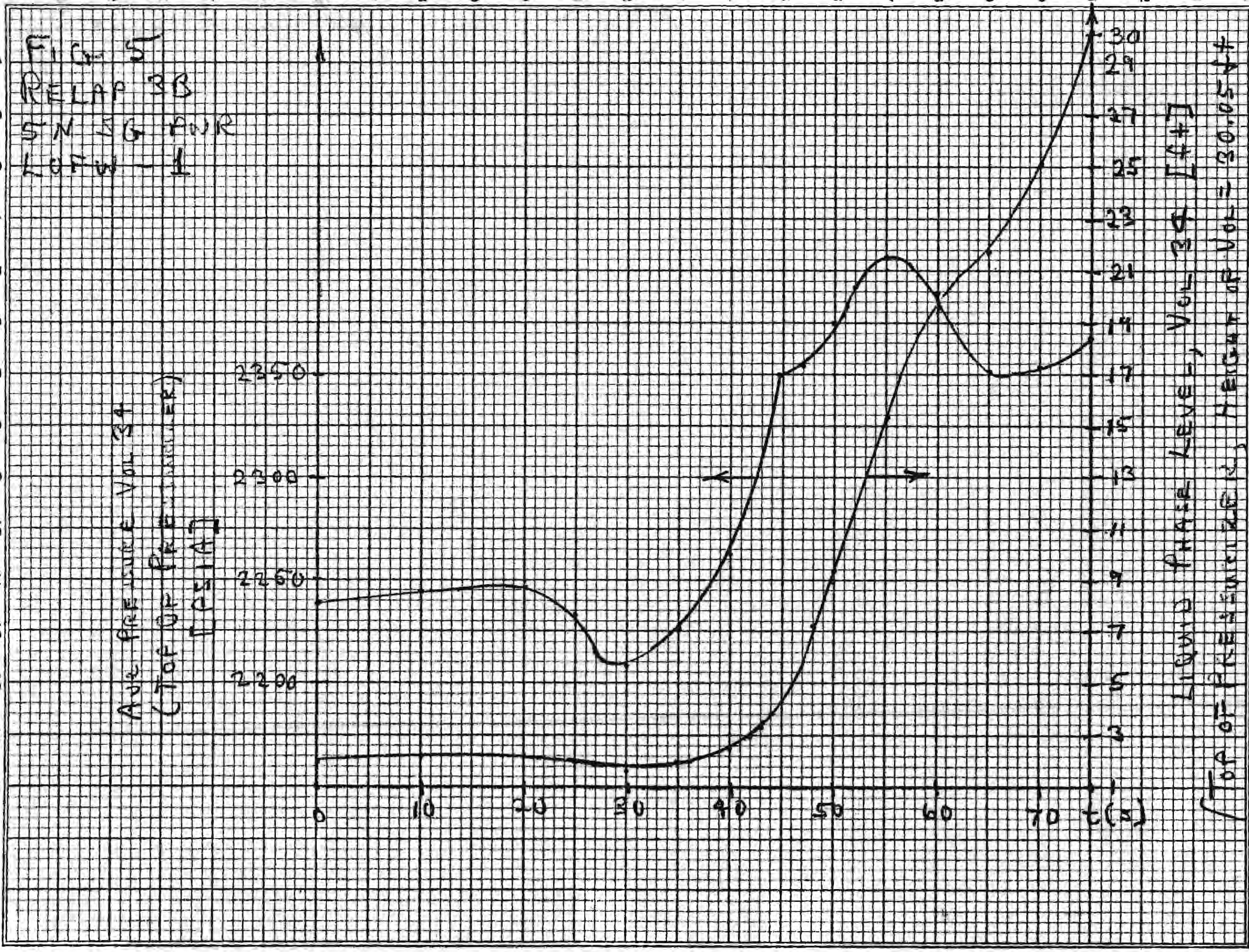
FIG 5
RELAP 3B
5N SG PWR
LOFW - 1
AVE PRESSURE VOL 34
(TOP OF PRESSURIZER)
[PSIA]
2200
2250
2300
2350
LIQUID PHASE LEVEL, VOL 34 [FT]
TOP OF PRESSURIZER, HEIGHT OF VOL = 30.05 FT
1
3
5
7
9
11
13
15
17
19
21
23
25
27
29
30
0
10
20
30
40
50
60
70
t(s)

Fig 2
RELATIVE
SATURATED FLUX
LOFW-1
1.0
0.9
0.8
0.1
0
10
20
30
40
50
60
70
t (s)
43
42
41
40
COOLANT DENSITY (lb/ft³)

GITNE-87/1

THE APPLICATION OF BENCHMARK INTEGRAL EXPERIMENTS TO THE DESIGN AND THE UNCERTAINTY ANALYSIS OF AN LDP-TYPE HETEROGENEOUS CORE

by John M. Kallfelz

Final Report for ORNL Subcontract 198-07802C
May 1987

ABSTRACT

Benchmark integral experiments are utilized to develop an adjusted data library for nuclear reactor calculations. This report describes the development and application of the adjusted cross-section and covariance library ORACLE-1. Methods and sources used to develop ORACLE-1 are documented, and techniques and results are discussed for application of this library to the design and uncertainty analysis of a large LMFBR core design from the Large Demonstration Plant (LDP) project.

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I. Introduction

As a result of inaccuracies of nuclear data, predictions of integral quantities for reactor design and performance have significant uncertainties [S-3]. The inaccuracies of predictions are revealed in comparisons with corresponding results of integral experiments in critical facilities and power reactors. There is no doubt that the deviations between theoretical and experimental results contain "information". There has been, however, considerable disagreement about how to utilize this information [0-1, 0-2, S-1], particularly in the United States. The attitudes on this topic are strongly influenced not only by technical and scientific reasons, but also by various emotional factors, so often significant in "big science" administration, such as protection of various programs when funding is limited, personal prejudices of funding administrators, etc. As discussed below, the resulting predominant positions and methods for utilization of this information are appreciably different in the U.S. and the European Community (EC). The various positions range from a generally negative attitude about data adjustment on the one hand to an optimistic attitude on the other. Two opposite positions which crystallized fairly early in the debate on this topic, and which are still reflected in some of today's programs are:

1. Integral experiments should merely be used for checking theoretical predictions and for finding indications of combined inaccuracies, caused by nuclear data as well as other sources of errors [Y-2].

2. Deviations between calculated and measured integral quantities should be utilized to adjust microscopic data, including their energy dependence, as described, for example, by group constant sets with 16-33 groups [C-5, H-3, C-6] or 2000 groups [C-7].

Between these two opposite positions a number of other possibilities have been investigated or proposed; e.g.

3. The use of all criticality measurements as a basis for an extensive interpolation procedure to find the criticality of other compositions [R-6].
4. Adjustment of group constant sets with a smaller number of groups (e.g., 4-5 groups) [R-2,B-5].

As with many other aspects in the development of the peaceful uses of nuclear energy, the most successful and sophisticated program for the use of experimental integral data for the reduction of design parameters is that of the French. Their CARNAVAL adjusted "formulaire" (including nuclear data), used for years for the design calculations for the PHENIX and SUPER PHENIX 1 reactors, has been markedly reliable for design parameter predictions. Presently a program at Masurca is supporting the design of SUPER PHENIX 2 [S-4]. Characteristic of the French program, clean configurations are chosen and various parameters are varied in a systematic manner to facilitate interpretation and data adjustment. [P-4]

Further, a step has been taken in France which has been studiously avoided in the U.S.; a Cadarache data bank (BD1) for LMFBR integral experiment data has been developed [R-1]. Such a well-documented and accessible data bank is crucial for the systematic utilization of integral data to improve the accuracy of design calculations. Presently BD1 is being used to perform adjustments on the new EC Joint Evaluated File [R-3] (JEF)[S-5].

The French have also pioneered in the systematic utilization of data from power reactors for nuclear data improvement [S-2]. Data from both PHENIX and SUPER PHENIX 1 are included in BD1 [R-1]. The sample and fuel pin irradiation experiments in PHENIX [D-1] are particularly useful.

"L' analyse des combustibles irradiés dans le coeur et les couvertures de PHENIX est une methode expérimentale extrêmement riche en informations pour le physicien, qu'il s'intéresse à l'evolution neutronique du combustible en reacteur ou à l'ensemble du cycle du combustible." [R-4]

While the information content of data from early PHENIX cycles has been investigated in the U.S. using time-dependent generalized perturbation theory, [K-3,K-4] these data have not been used for data adjustment.

Recent work [P-5] is indicative of a growing recognition of cross section experimentalists in the U.S. of the need for a systematic inclusion of integral experiment information in the calculation of integral parameters.

Compared to the European Community program, the use and development of adjusted data libraries in the U.S. has been modest indeed. Such adjustment conflicted with the basic philosophy of the ENDF development. While EC researchers have openly embraced the data adjustment methods, the

resistance to such methods in the U.S. was extraordinary. Defense of turf played a not insignificant role in the development and maintenance of this resistance. The random way in which life is appreciably influenced by factors that should be per se insignificant is illustrated by the fact that much of the resistance to adjustment was caused by the experience of an influential DOE administrator. At a decisive period of his career this administrator was delegated to Britain, where (according to a grapevine anecdote which appears substantiated) because of the questionable adjustment methods employed by a group with whom he worked, he developed a significant prejudice against such methods.

Of particular interest in the French program is the use of the residual (after data adjustment) differences between experimental (E) and calculated (C) integral values to further improve (reduce their uncertainty) calculated values for reference design configurations [P-4]. "Each configuration [is characterized by] an indicator, which, for spectrum-dependent integral parameters, has been defined as a spectrum-dependent parameter value r" [P-4]. Using the r value for the reference power reactor configuration, interpolation yields residual E-C values, and their associated uncertainties, for reactor design parameters calculated with adjusted data [P-4].

At any rate, one of the results of the very modest U.S. adjustment program is ORACLE-I, [Y-1] an adjusted library based on ENDF/B-V, developed at Oak Ridge National Laboratory. The acceptance and use of this library in the design community has frankly been quite limited. It is hoped that a documentation of the methods and sources used to develop ORACLE, along with

some results calculated with ORACLE for a large LMFBR design from the Large Demonstration Plant (LDP) Project, will promote wider usage and further improvement of this data library. The purpose of this paper is such a documentation.

II. GENERAL DESCRIPTION OF THE LDP-TYPE DESIGN*

The key design parameters for a large heterogeneous LMFBR of the 1000 MWe class were specified as part of the DOE-sponsored Large Demonstration Plant (LDP) project. The design objectives and criteria for the LDP project (originally called the Conceptual Design Study [CDS] project [D-2]) emphasized the following: a) high reliability, b) near term design features (components that can be developed within five years), c) sufficiently low sodium void worth to preclude hypothetical core disruptive accidents from consideration as design basis accidents, d) breeding of fissile fuel at a rate equivalent to a compound system doubling time of twenty years or less, and e) allowances for the future accommodation of advanced fuels.

II. A. Model

The LDP-type reactor model employed was the General Electric design, [K-2, M-1] a 2540 MW_{th}, plutonium-uranium oxide fueled heterogeneous reactor with a driver fuel, blanket, shield and control assembly layout as

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*Principal author of this section: C.L. Cowan, General Electric Co., Sunnyvale, California

shown in Fig. 1. The tightly coupled core has a lattice pitch of 15.062 cm and an active fuel height of 101.60 cm. A listing of the principal design parameters and the zonewise material compositions for the reference system is given in Table I.

The driver fuel and blanket specifications for the LDP-type design provide a trade-off between the requirements for a system with low energetics (i.e., a low positive sodium void reactivity), and the requirements for a doubling time of less than about 20 years.

The core layout, shown in Fig. 1, was specified to minimize the peak radial power throughout the equilibrium operating cycle based upon a single fissile enrichment for the supplied driver fuel. The twelve control rods in the outer driver fuel zone are also utilized to shape the power profile throughout the irradiation cycle.

The fuel management scheme during the equilibrium operating cycle was established on the basis of the following assumptions:

1. The supplied fuel for the driver fuel regions is discharged plutonium from a water reactor system. The plutonium isotopic ratios for the supplied fuel are:

Pu-238	0.00997
Pu-239	0.67272
Pu-240	0.19209
Pu-241	0.10127
Pu-242	0.02395
	1.00000

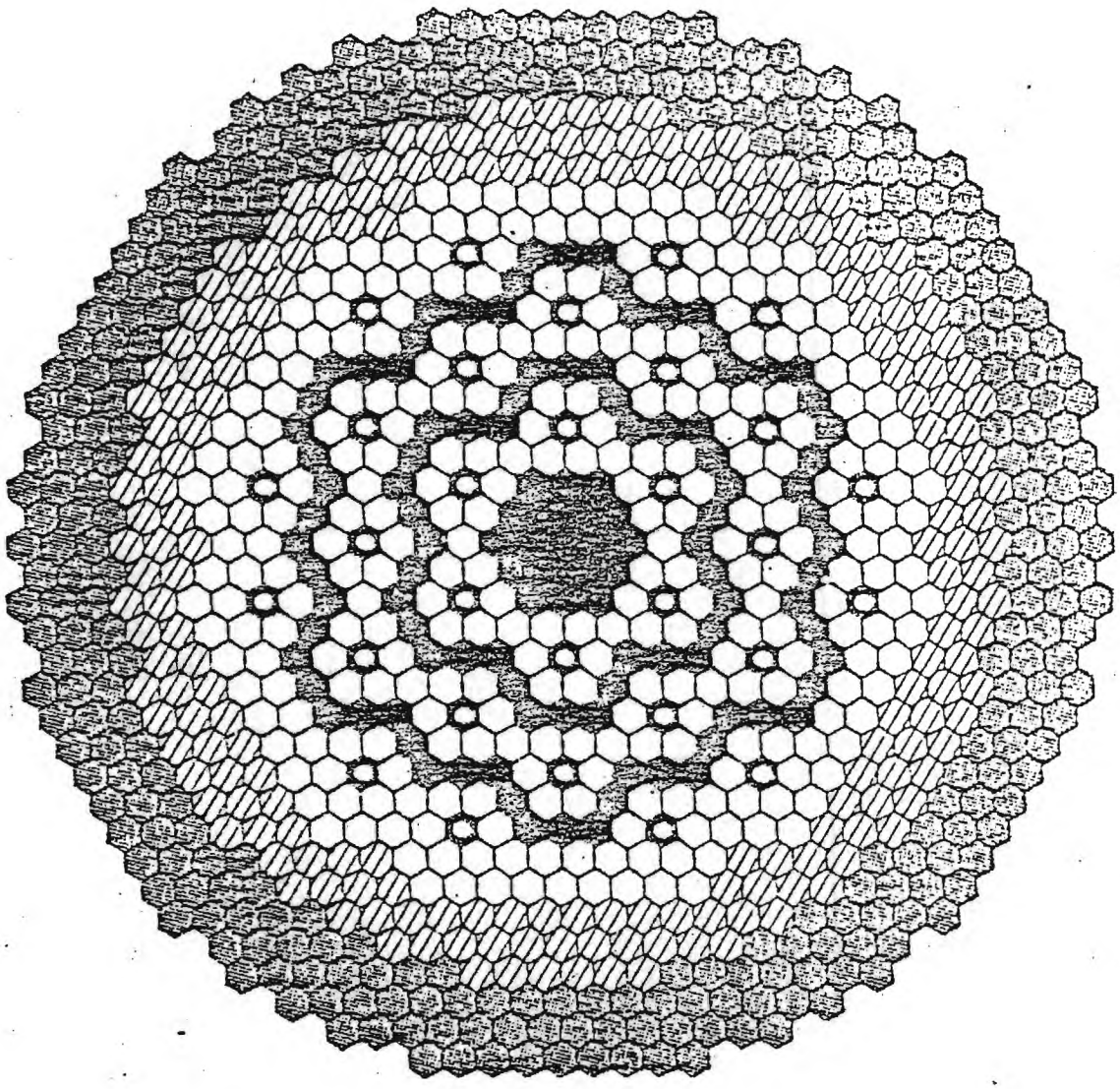


FIGURE 1. CORE LAYOUT FOR THE LDP-TYPE REFERENCE DESIGN

2. The supplied fuel for all blanket regions is depleted uranium containing 0.2 atom percent U-235.
3. The plant capacity factor is 0.8.
4. All driver and blanket fuel assemblies are irradiated at a fixed location based upon a scatter reload scheme. The driver and inner blanket fuel assemblies remain in the reactor for two years. The inner, middle, and outer radial blanket assemblies remain in the reactor for 3, 4, and 5 years, respectively.

A nominal twelve month refueling interval was specified for the reference system. Based upon the fuel management plan described above, the average exposure of the discharge driver fuel was computed to be 58,800 MWD/MT.

The control requirements for the LDP-type design are satisfied with two independent and diverse systems. The primary control system consists of 3 rods from the inner control ring, 6 rods from the middle control ring and all 12 rods from the outer control ring (a total of 21 rods). This system is utilized to maintain criticality throughout the operating interval, and to shut the reactor down from hot-full power to zero power at ambient temperatures with the highest worth control rod stuck. The remaining 9 rods (see Fig. 1) comprise the secondary control system. This system is required to shut the reactor down from hot-full power to hot standby conditions at zero power with the highest worth control rod stuck.

II. B. Analysis Methods

The design calculations for the LDP-type design were performed by C. L. Cowan and R. Protsik of General Electric [K-2, M-1], using ENDF/B-V data and the analysis methods described in this section. Steady-state flux solution calculations were performed using diffusion theory in the two-dimensional neutron transport code SN2D. The computations were carried out in R-Z and X-Y geometries which were linked by the axial buckling terms. In specifying the R-Z design the driver fuel, blanket and control regions were modeled into annular rings with the region widths adjusted to flatten the radial power distribution. In this approach the planar power distributions and control worths were determined using the X-Y configuration, whereas, the fuel inventory and global design parameters (e.g., breeding ratio, Doppler coefficient, etc.) were determined from the R-Z representation. All fuel cycle calculations were carried out using the fuel management and burnup code FUMBLE [C-2].

The region wise microscopic cross sections for the LDP-type design were generated by utilizing the shielding factor (f-factor) methodology [K-5] as incorporated in the TDOWN-IV code [P-3]. All data processing calculations started with a 70-group generalized data file based upon ENDF/B-V, and included a cell heterogeneity correction (i.e., an explicit treatment of the pin and assembly geometry). The 70-group data file was collapsed to few-groups using the fluxes from several one-dimensional flux solution computations. In general, the performance parameters for the LDP-type design were computed using a 6-group cross section file. The mass balances and breeding ratio, though, were adjusted on the basis of a final

22-group calculation. The sodium void and Doppler coefficients were also computed using a 22-group cross section file.

II. C. Design Characteristics

The key performance parameters for the LDP-type reference design are briefly described in Sections II. C. 1. - II. C. 4.

II. C. 1. Power Distribution

The power distribution during the operating interval was found to be relatively flat with only a small shift in the normalized power densities from the inner core zones to the outer core zones as the fuel was irradiated. Note that the outer 12 control rods (see Fig. 1) are utilized to help shape the power profile. The region wise power fractions for the LDP-type design were computed at the beginning and end of the equilibrium cycle (BOEC and EOEC), and are given in Table II.

The power split is characterized by the significant fraction of the total power which is generated in the inner blanket assemblies. The buildup of fissile plutonium in these assemblies is an important factor in reducing the burnup reactivity swing during the operating interval.

The nominal peak linear power during the equilibrium cycle was computed to be 12.6 kw/ft at BOEC. The peak power corresponds to a peak-to-average power ratio at BOEC of approximately 1.49. The location of the peak power assembly in the heterogeneous design was found to be dependent

upon the fuel management plan, and may occur in either the middle or outer driver fuel zones.

Based upon the fuel management scheme described in Sec. II. A., the average exposures for the driver fuel and inner blanket assemblies (excluding the axial blanket extensions) were calculated to be 58,800 MWD/MT and 11,400 MWD/MT respectively. The peak driver fuel burnup was found to be about 87,000 MWD/MT.

II. C. 2. Fuel Inventory, Breeding Ratio and Doubling Time

The fissile plutonium inventories for the LDP-type design are listed, in Table III, at BOEC and EOEC. The fissile gain for the equilibrium cycle corresponds to a breeding ratio of 1.35. The compound system doubling time for the reference design was computed to be 16.4 years based upon a 1 year out-of-pile reprocessing time, and a combined reprocessing and fabrication fissile loss of 1.0 percent.

The plutonium fissile enrichment for the supplied fuel at the beginning of the equilibrium cycle was determined to be 17.5 percent. Because the fissile enrichment is 40 to 50 percent higher than that for a comparable homogeneous system, the driver fuel conversion ratio is significantly lower than that for the homogeneous design (i.e., an internal conversion ratio of 0.65 for the heterogeneous design versus approximately 0.95 for the homogeneous design). The fissile depletion in the driver fuel is compensated by the fissle buildup in the inner blankets so that the net reactivity change during the operating cycle is small.

II. C. 3. Safety Coefficients

Sodium void calculations for the LDP-type design were performed by removing the flowing sodium (e.e., sodium inside the hex can) from the reactor regions of interest. The results of void calculations at BOEC and EOEC are listed in Table IV. The positive void reactivity for the driver fuel plus axial blanket extensions satisfied the design objectives for a value of less than $2.5. However, some care should be taken in predicting the impact of the void reactivity on the overall system energetics, because of the large uncertainties in the propagation of the sodium boiling phenomena in the driver fuel and inner blanket assemblies.

Doppler calculations were carried out at EOEC on the basis of direct flux solution calculations in which the fuel temperature was increased from 1500 to 2100°K. The results of the Doppler calculations for the sodium-in and sodium-out reference cases are listed in Table V. The axial and radial blankets contribute an additional 3 to 4 percent to the total Doppler effect. Results of the LDP-type calculations have also indicated that the BOEC Doppler is from 6 to 8 percent higher than the EOEC value. Thus, the positive sodium void reactivity is the greatest at the EOEC, whereas, the Doppler is the lowest at EOEC.

II. C. 4. Reactivity Requirements and Control Worth

The primary control system must be capable of compensating for the system reactivity requirements which include the hot full-power to zero-

power temperature defect, a $1.00 shutdown margin, the excess reactivity for a 1 year irradiation cycle, and design uncertainties which impact the system criticality. These reactivity requirements, including the uncertainty margins, have been computed for the reference design to be approximately $13.6. An investigation of the worth of the primary system has indicated that the reactivity requirements can be satisfied by a 21-rod assembly (including one stock rod) in which the boron carbide poison is 30 percent enriched in B-10.

The secondary control system must be capable of compensating for the system reactivity requirements which include the hot full-power to zero power (hot standby) temperature defect, and a $1.0 shutdown margin. These reactivity requirements, including the uncertainity margins in the temperature defect, have been computed to be $5.3. An investigation of the worth of the secondary control system has indicated that the reactivity requirements can be satisfied by a 9-rod assembly (including one stuck rod) in which the boron carbide poison is 80 percent enriched in B-10.

TABLE I
DESIGN CHARACTERISTICS FOR AN LDP-TYPE
HETEROGENEOUS REACTOR

Design Parameter	Specifications Driver Assembly/Blanket Assembly
Core Height (cm)	101.6
Axial Blanket Thickness (cm)	35.56
Duct Pitch (cm)	15.062
Duct Gap (cm)	0.5588
Duct Wall Thickness (cm)	0.3556
Pins/Assembly	271/127
Pin Outer Diameter (cm)	0.6985/1.1176
Clad Thickness (cm)	0.03683/0.03556
Wire Wrap Pitch (cm)	30.48/15.24
Wire Diameter (cm)	0.12192/0.08128
Edge Ratio	1.0/1.0
Fuel Smear Density (% theoretical)	86.5/93.3

VOLUME FRACTIONS

Material	Driver Fuel	Inner & Radial Blankets	Radial Shield	Primary Control In/Out	Secondary Control In/Out
Fuel ($Pu-UO_2$)	0.4243	0.5577	-	-	-
Structure (D9)	0.2111	0.1713	0.7884	0.3400/0.0887	0.2793/0.1431
Coolant (Na)	0.3646	0.2710	0.2116	0.3198/0.9113	0.3632/0.8569
Control	-	-	-	0.3402/0.0	0.3575/0.0

Table II

Regionwise Power Fractions for the LDP-type Design

	Power Fractions	
	BOEC	EOEC
Driver Fuel	0.859	0.794
Internal Blankets	0.081	0.129
Axial Blanket Extensions of Driver Fuel	0.017	0.024
Radial Blankets	0.038	0.050
Shields and Control	0.005	0.003
	1.000	1.000

Table III

Fissile Plutonium Inventories for the LDP-type Design

	Fissile Inventory (kg of Pu-239 plus Pu-241)	
	BOEC	EOEC
Driver	3605.0	3336.8
Internal Blankets	152.1	428.2
Axial Blanket Extensions of Driver Fuel	55.4	162.4
Radial Blankets	219.0	386.0
Total	4031.5	4313.4
Fissile Gain	281.9	

Table IV

Results of Sodium - Void Calculations for the LDP-type Design

	Sodium Void Reactivity ($)*	
Voided Regions	BOEC	EOEC
Driver Fuel	2.07	2.61
Driver Fuel Plus Axial Blanket Extensions	1.85	2.39
Driver Fuel, Inner Blankets Plus Axial Blanket Extensions	2.80	3.43

*One dollar ($1.0) is equal to 0.36% Δk.

Table V

Results of Doppler Calculations at EOEC for the LDP-Type Design

	Doppler Coefficient at EOEC ($T\Delta k/\Delta T$)
Sodium-In	
Driver Fuel	- 0.0053
Internal Blankets	- 0.0032
Sodium-Out	
Driver Fuel	- 0.0043

III. Sensitivity Studies of Core Performance Parameters

In this section we discuss the methods and model used to determine the sensitivities of various performance parameters to cross section changes, as well as the results thereof. The resulting relative "sensitivity coefficients" (SC) are defined for an integral parameter R as:

$$SC_{\sigma}^{R} = \frac{\sigma}{R} \cdot \frac{\partial R}{\partial \sigma} \qquad \text{(III-1)}$$

where σ is a function of energy group, nuclide, and reaction.

The ORACLE-I cross-section set, discussed in Section IV, was used for all the results reported in this section.

III. A. Theory Employed

The calculation of sensitivity coefficients by generalized perturbation theory (GPT) has been extensively discussed in the literature [M-2, G-3, C-3], and a summary of the relevant equations are contained in References [M-2] and [C-3]. The following discussion applies to parameters R which are functions of the real flux, $\vec{\phi}(\vec{r})$. For the case of the sodium-void reactivity we did not employ GPT; as discussed in section III. C. 2, we used a method which is equivalent to the later-developed "equivalent generalized perturbation theory" (EGPT) for reactivity worths [G-1].

The relative change in an integral ratio R for a cross-section perturbation consists of two components, due to a "direct" and "indirect" effect. The direct effect arises because of changes in the cross sections which appear in the definition of R, and is calculated with a trivial

expression which uses the unperturbed value of $\phi(\vec{r})$. The indirect effect, caused by changes in $\phi(\vec{r})$ due to changes [δB] in the Boltzmann operator associated with the nuclear data changes, can be calculated with the methods of generalized perturbation theory with the following multi-group expressions.

$$\frac{\delta R}{R}\Big|_{\text{indirect}} \sim \int \vec{\Gamma}^{*}[\,\delta B]\,\vec{\phi}\; d\,\vec{r} \qquad \text{(III-2)}$$

$\vec{\Gamma}^{*}(\vec{r})$ is the generalized adjoint function, whose component $\Gamma_j^{*}(r)$ gives the importance of neutrons at $\vec{r}$ in energy group j to the associated reaction rate ratio, R. Γ_j^{*} satisfies an equation identical in form to the normal adjoint equation for ϕ^{*}_j, except for the presence of a fixed source term [M-2, C-3].

The influence of a cross-section change on an integral parameter depends on the method used for "k-reset,"[M-2] i.e., the change in the reactor configuration made to keep k_{eff} invariant by compensating for the change in k_{eff} caused by the initial cross-section change. The "k-reset" mechanism used in this study was a variation in the driver zone "plutonium enrichment," i.e., (total Pu)/(total heavy metal). This mechanism is felt most appropriate to represent the characteristics of an equilibrium cycle core, for which fuel enrichment can be adjusted to compensate for first-cycle excess reactivity above (or below) the calculated value.

The resulting sensitivity coefficients are defined for change in a specific group cross section with the approximation that all self-shielding effects were fixed. For all reported sensitivities, σ_t was treated as

dependent, i.e., a perturbation in one of the partial cross sections had an associated $\delta\sigma_{tr}$. This convention will not influence the integral parameter uncertainties, to be discussed in Section V, if consistent cross-section covariance data is used in the determination of these uncertainties.

III. B. Model and Methods

For all the sensitivity coefficient calculations, the R-Z reactor model described in Reference K-2 was used. This model is not identical to those employed to determine the absolute values reported in Section II, but is a model characteristic of the middle of equilibrium cycle (MOEC) conditions in the LDP reactor. Control is virtually all withdrawn, with only the outer control bank inserted slightly to achieve criticality for the nuclear data set used in Reference K-2.

The ORACLE-I cross-section set and diffusion theory were used to determine the sensitivity coefficients utilizing VENTURE [V-1] and the DEPTH-CHARGE [W-2] sensitivity code. Thus, for our sensitivity coefficient calculations we generally used the same geometry (R-Z) and method (diffusion theory) used to determine the design parameters reported in Section II.

III. C. Parameters Studied

III. C. 1. k_{eff} and Breeding Ratio

These parameters have been investigated in many previous sensitivity studies, [M-2, G-4, K-2, M-4, K-1] and we have used the same basic techniques as those employed therein. It should be noted that we have used the following breeding ratio (BR) definition,

$$BR = \frac{\int_{\text{reactor}} \sum_j \Sigma_{1,j}(\vec{r})\,\phi_{\theta j}(\vec{r})\,d\vec{r}}{\int_{\text{reactor}} \sum_j \Sigma_{2,j}(\vec{r})\,\phi_j(\vec{r})\,d\vec{r}} \qquad \text{(III-3)}$$

where Σ_1 and Σ_2 are the fertile capture and fissile absorption cross sections, respectively. Thus, our results are for a static MOEC breeding ratio, rather than the cycle-averaged value reported in Section II.

III. C. 2. Sodium-Void Reactivity

Sodium-void reactivity sensitivity coefficients have been investigated by GPT in many studies [C-3, G-4, G-5, G-6, H-1]. However, these investigations were performed with 1D models, which can give poor results for the "diffusion" term of perturbation expressions due to inaccuracies in calculated flux and adjoint gradients [K-1]. Such diffusion terms can be appreciable for sodium voiding of reactor regions.

The accuracy of sodium-void reactivity calculations can be influenced by various methods and models which are more accurate than the analysis methods discussed in Section II, e. g., transport theory, 3D model, and streaming effect corrections [K-6, B-3, B-4, S-3]. For our calculations we used the same methods as those used to determine the values reported in

Section II, i.e., multigroup 2D diffusion theory, with group cross sections generated for the base and voided cases. We examined the case of voiding of the flowing sodium (about 81% of the total) in the driver zones. Using the ORACLE-I cross-section set, our calculated value for the associated reactivity is 572 pcm, or about \$1.6 assuming that one dollar is equal to 360 pcm.

Since sodium-void reactivity values are composed of several compensating terms, e. g., a positive spectral term and a negative leakage term, calculated total values are sensitive to minor changes in cross-section values. This is the primary reason for the difference between our sodium-void reactivity value of \$1.6 at MOEC, and the ENDF/B-V average value of about \$2.3 between the BOEC and EOEC values for the same voiding reported in Section II. Part of the difference is due to the different fuel temperatures (1200°K and 1500°K) assumed for the ENDF/B-V and ORACLE-I group cross sections, respectively. Another contributor is the difference in the ENDF/B-V and ORACLE-I data base.

To determine the sodium-void sensitivity coefficients, GPT was not necessary since these parameters can be determined from the difference in results from two "normal" perturbation theory calculations, e.g.:

$$SC_{\sigma}^{\text{Na-void}} = \frac{1}{R}\left[-\frac{1}{k_1} SC_{\sigma}^{k1} + \frac{1}{k_2} SC_{\sigma}^{k2} \right] \qquad \text{(III-4)}$$

where R is the sodium-void reactivity, and k_1 and k_2 signify k_{eff} for the base and voided cases, respectively. For perturbation of a single cross-section set the sensitivities calculated by the above expression and by GPT are identical. This method has recently been further developed and investigated in the development of "equivalent GPT" (EGPT) [G-1].

In considering our results for the sodium-void relative sensitivity coefficients, which are proportional to $\delta R/R$ for a given perturbation [see Eq. (III-1)], one should consider that the associated absolute R value is lower than the sodium-void reactivities given in Section II. For some cases the sodium-void reactivity is nearly zero and the significance of the relative change $\delta R/R$ is not great. For these cases the absolute sensitivity coefficient, which involves the absolute value of δR, is more meaningful.

III. D. Sensitivity-Coefficient Results

III. D. 1. k_{eff} and Breeding Ratio Sensitivities

The general characteristics of these sensitivities, including the influence of the plutonium enrichment "k-reset" mechanism, have been discussed in previous references [M-2, K-2, M-4]. For these parameters, selected total (energy-integrated) sensitivities are given in Table III-1.

In general, these sensitivity coefficients are within about fifteen percent of those reported in Ref. K-2, which were calculated with ENDF/B-IV data for the same reactor model. Some of the smaller total scatter sensitivities (sum of the elastic and inelastic coefficients), i.e., those with absolute values less than 0.001, show appreciable relative differences from our previous results [K-2], but this is not surprising considering the characteristic dependence of scattering perturbation results on the cross-section data used.

The largest sensitivities are for heavy metals, and the five largest total sensitivities for k_{eff} and BR for our present results and for a homogeneous reactor [M-4] are for the same reactions and have roughly the same values for the two reactions.

It should be noted that, as discussed further in Section V., the largest sensitivities do not necessarily indicate the most significant data for the performance parameter uncertainties, which are also dependent on the uncertainties of the nuclear data. An obvious example of this characteristic is ν, which has large sensitivities but small uncertainties.

III. D. 2. Sodium-Void Reactivity Sensitivities

Selected total relative sensitivity coefficients for the sodium-void reactivity discussed in Section III. C. 2. are presented in Table III-2. As expected, these sensitivities are large for nuclear data which have large k_{eff} sensitivity coefficients. Besides various heavy metal reactions, the scatter sensitivities for sodium, oxygen, and iron are large. The values of these sensitivities for the scattering cross sections were strongly influenced by the associated $\delta\sigma_{tr}$ deriving from the convention that σ_t is dependent, as discussed in Section III. A.

Fig. III-1 shows an example of the energy dependence of a sodium-void sensitivity, specifically that for U-238 σ_c. The positive and negative influence of increased capture in the low and high energy ranges, respectively, is principally due to the influence of this perturbation on the energy variation of the adjoint flux, which strongly influences the "spectral" or "scattering" component of the sodium-void reactivity [H-2].

Table III-1

Selected Total Relative Sensitivity Coefficients for k_{eff} and BR of the LDP Core Using the ORACLE-I Data Set

Nuclide	Reaction	k_{eff} Sensitivity	BR[a] Sensitivity
U235			
	ν	9.7 E-3	4.0 E-3
	σ_f	6.8 E-3	-1.3 E-2
	σ_c	-8.5 E-4	-5.6 E-3
U238			
	ν	1.3 E-1	1.7 E-2
	σ_f	7.8 E-2	1.0 E-2
	σ_c	-2.2 E-1	7.7 E-1
	σ_{el}	2.0 E-2	-1.8 E-2
	σ_{inel}	-5.2 E-2	1.8 E-2
Pu239			
	ν	6.9 E-1	-1.2 E-2
	σ_f	4.9 E-1	-6.6 E-1
	σ_c	-5.2 E-2	-1.7 E-1
	σ_{el}	1.6 E-3	-3.3 E-3
	σ_{inel}	-2.6 E-3	-1.0 E-4
Pu240			
	ν	5.2 E-2	-2.9 E-3
	σ_f	3.5 E-2	-2.1 E-3
	σ_c	-1.7 E-2	5.1 E-2
	σ_{el}	5.2 E-4	-1.2 E-3
	σ_{inel}	-1.3 E-3	b
Pu241			
	ν	1.2 E-1	-5.7 E-3
	σ_f	8.5 E-2	-1.1 E-1
	σ_c	-5.9 E-3	-1.8 E-2
	σ_{el}	1.7 E-4	-3.8 E-4
	σ_{inel}	-6.3 E-4	b

Table III-1 (cont)

Pu242			
	ν	4.5 E-3	-2.7 E-4
	σ_f	3.1 E-3	-2.0 E-4
	σ_c	-1.7 E-3	4.8 E-4
	σ_{el}	6.8 E-5	1.5 E-4
	σ_{inel}	-1.9 E-4	b
NA-23			
	σ_c	-1.4 E-3	-3.8 E-4
	σ_{el}	4.9 E-3	-3.5 E-2
	σ_{inel}	-7.4 E-3	-2.1 E-4
O-16			
	σ_{el}	-2.8 E-2	-1.1 E-2
	σ_{nel}	-2.6 E-4	b
Fe			
	σ_c	-1.1 E-2	-4.4 E-3
	σ_{el}	1.2 E-2	-2.4 E-2
	inel	-1.8 E-2	-7.5 E-4
Ni			
	σ_c	-5.4 E-3	-2.8 E-3
	σ_{el}	4.4 E-3	-1.4 E-2
	σ_{inel}	-3.5 E-3	b
Cr			
	σ_c	-5.5 E-3	-2.3 E-3
	σ_{el}	3.9 E-3	-9.7 E-3
	σ_{inel}	-3.8 E-3	-2.3 E-4
Mo			
	σ_c	-3.9 E-3	-1.7 E-3
	σ_{el}	5.8 E-4	-9.2 E-4
	σ_{inel}	-8.9 E-4	b
C	σ_{el}	1.5 E-4	-6.1 E-4

[a]Convert BR sensitivities to k-reset values by adding 1.79 x k_{eff} sensitivity.

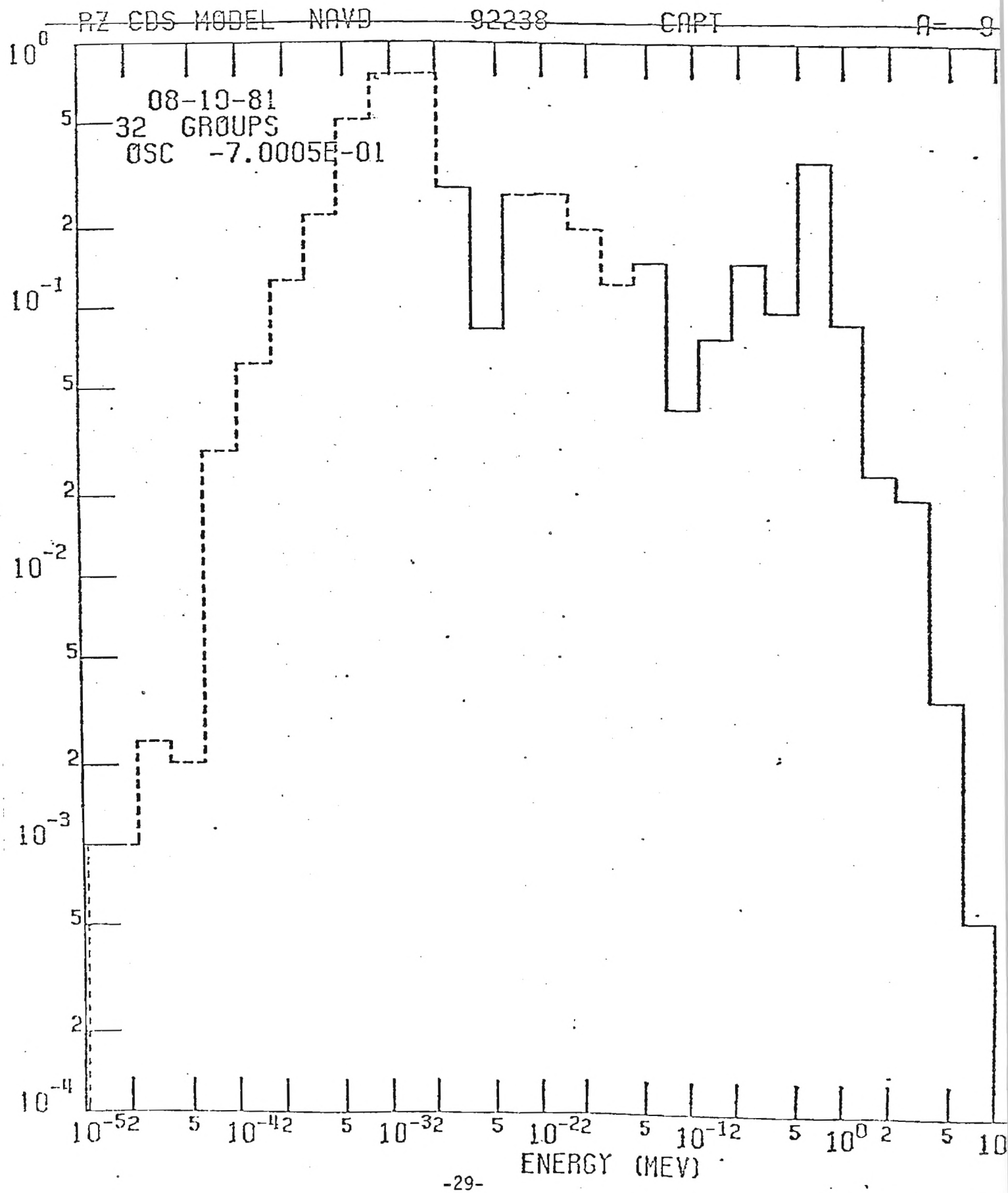
[b]Absolute sensitivity coefficient value $< 1 \cdot 10^{-4}$.

Table III-2

Selected Total Relative Sensitivity Coefficients for the Sodium-Void Reactivity of the CDS Core Using the ORACLE-I Data Set (no k-reset)

Nuclide	Reaction	Sensitivity	Nuclide	Reaction	Sensitivity
U-235			Pu-239		
	ν	-1.1 E-2		ν	-1.7
	σ_f	-2.9 E-3		σ_f	-9.9 E-1
	σ_c	6.5 E-3		σ_c	9.0 E-1
	σ_{el}	2.6 E-4		σ_{el}	1.6 E-2
	σ_{inel}	-1.1 E-3		σ_{inel}	-3.6 E-2
U-238			Pu-240		
	ν	1.1		ν	2.9 E-1
	σ_f	6.5 E-1		σ_f	1.9 E-1
	σ_c	9.8 E-1		σ_c	2.8 E-1
	σ_{el}	1.4 E-1		σ_{el}	3.7 E-3
	σ_{inel}	-6.4 E-1		σ_{inel}	-1.5 E-2
Pu-241			Pu-242		
	ν	-7.1 E-1		σ_f	2.9 E-2
	σ_f	-4.5 E-1		σ_f	1.9 E-2
	σ_c	8.5 E-2		σ_c	2.5 E-2
	σ_{el}	1.3 E-3		σ_{el}	4.7 E-4
	σ_{inel}	-8.1 E-3		σ_{inel}	
Na-23			O-16		
	σ_c	1.1 E-1			
	σ_{el}	-1.8 E-1		σ_{el}	-5.6 E-1
	σ_{inel}	6.5 E-1		σ_{inel}	-6.6 E-3
Fe			Ni		
	σ_c	1.0 E-1		σ_c	2.1 E-2
	σ_{el}	5.3 E-3		σ_{el}	-6.5 E-2
	σ_{inel}	-2.2 E-1		σ_{inel}	-4.7 E-2
Cr			Mo		
	σ_c	3.6 E-2		σ_c	4.8 E-2
	σ_{el}	3.0 E-2		σ_{el}	5.7 E-3
	σ_{inel}	-5.0 E-2		σ_{inel}	-1.2 E-2
C	σ_{el}	-4.8 E-3			

Figure III-1. Relative Sensitivity Coefficient per Unit Lethargy for U-238 σ_c. (Dotted and solid lines are for positive and negative values.)



IV. The ORACLE-I Adjusted Library

(Outline from J. Wagschal [telecom], who has the responsibility for writing this section.)

A. Brief introduction - Rationale behind development.

B. Basic formulas for adjustment procedures.

C. -Experimental responses selected.

D. -Calculated values for responses.

-Procedures for calculations.

-Codes used for calculations.

-Calculation Bias Factors.

(Het-hom correction, e.g.).

E. Discussion of uncertainties, including methods uncertainties.

-Source of uncertainties.

F. Checking data (experiment) consistency before performing adjustment.

G. Methods for testing of adjusted library.

-Check values calculated w/adjusted set.

(How close to exp. values; how much uncertainty reduced.)

H. Link to following section.

V. Uncertainties of Calculated Performance Parameters

To determine the uncertainty of an integral parameter R, one can calculate its variance (VAR):

$$VAR(R) = \sum_{i,j} \frac{\partial R}{\partial \sigma_i} \frac{\partial R}{\partial \sigma_j} COV(\sigma_i, \sigma_j) \qquad (V\text{-}1)$$

where $\partial R/\partial \sigma$ and $COV(\sigma_i, \sigma_j)$ denote sensitivities and covariances of various cross sections, respectively. [K-2] To obtain a reliable estimate of the standard deviation $SD(R) = [VAR(R)]^{1/2}$, Eq. (V-1) should include all data which have a significant impact on R.

Note also that the covariance of two integral parameters R_1 and R_2 is given by

$$COV(R_1, R_2) = \sum_{i,j} \frac{\partial R_1}{\partial \sigma_i} \frac{\partial R_2}{\partial \sigma_j} COV(\sigma_i, \sigma_j), \qquad (V\text{-}2)$$

while the correlation $COR(R_1, R_2)$ between R_1 and R_2 is expressed as

$$COR(R_1, R_2) = COV(R_1, R_2) / [SD(R_1)\ SD(R_2)]. \qquad (V\text{-}3)$$

For the results reported in this section, the sensitivities and covariances $COV(\sigma_1, \sigma_2)$ used are based on the ORACLE-I adjusted cross section library. As mentioned in Sec. III. A., for all reported sensitivities, the partial reaction cross sections were taken to be the independent variables, and the transport

and total cross sections were treated as dependent variables. Thus for the diffusion theory calculations, a perturbation in a partial cross section had an associated perturbation in the transport cross section. For consistency the same convention must be used for the covariance data $COV(\sigma_i, \sigma_j)$ in Eq. (V-2).

Standard deviations for k_{eff} are obtained by using Eq. (V-1), and the resulting values are presented in Table V-1. The value of 0.4% shown in this table has been obtained by using ORACLE-I based nuclear data sensitivities and covariances of Fig. V-1. For the sake of comparison, Table V-1 also presents previously reported [M-1, K-1] standard deviations for k_{eff}. Thus, the value of 3.2% was obtained [K-1] for the same LDP-type reactor model, but by using unadjusted ENDF/B-IV data and sensitivities. The marked difference between this "unadjusted" value and the "adjusted" value of 0.4% is largely due to the effect of including results of integral experiments.

Also shown in Table V-1 are the previously reported [M-1] standard deviations for k_{eff} for the homogeneous LCCEWG-LMFBR reactor. The value of 3.1% was obtained by using sensitivities and covariances based on ENDF/B-IV data.* Methods and modeling biases, [M-1] and integral experiment results were omitted in this case. When the respective biases and integral experiment result were included, the adjustment reduced the standard deviation from 3.1% to 0.5%. Note that, although only thirteen benchmark integral experiments (i.e., ZPR-6/7 k_{eff}, $^{28}f/^{49}f$, $^{28}f/^{49}f$, $^{28}c/^{49}f$; ZPR-6/6A k_{eff}, $^{28}f/^{25}f$, $^{28}c/^{25}f$; and ISNF $^{49}f/^{25}f$, $^{28}f/^{25}f$) were included in the adjustment procedure [M-1] for the LCCEWG-LMFBR, the reduction in the $SD(k_{eff})$ from 3.1% to 0.5% is

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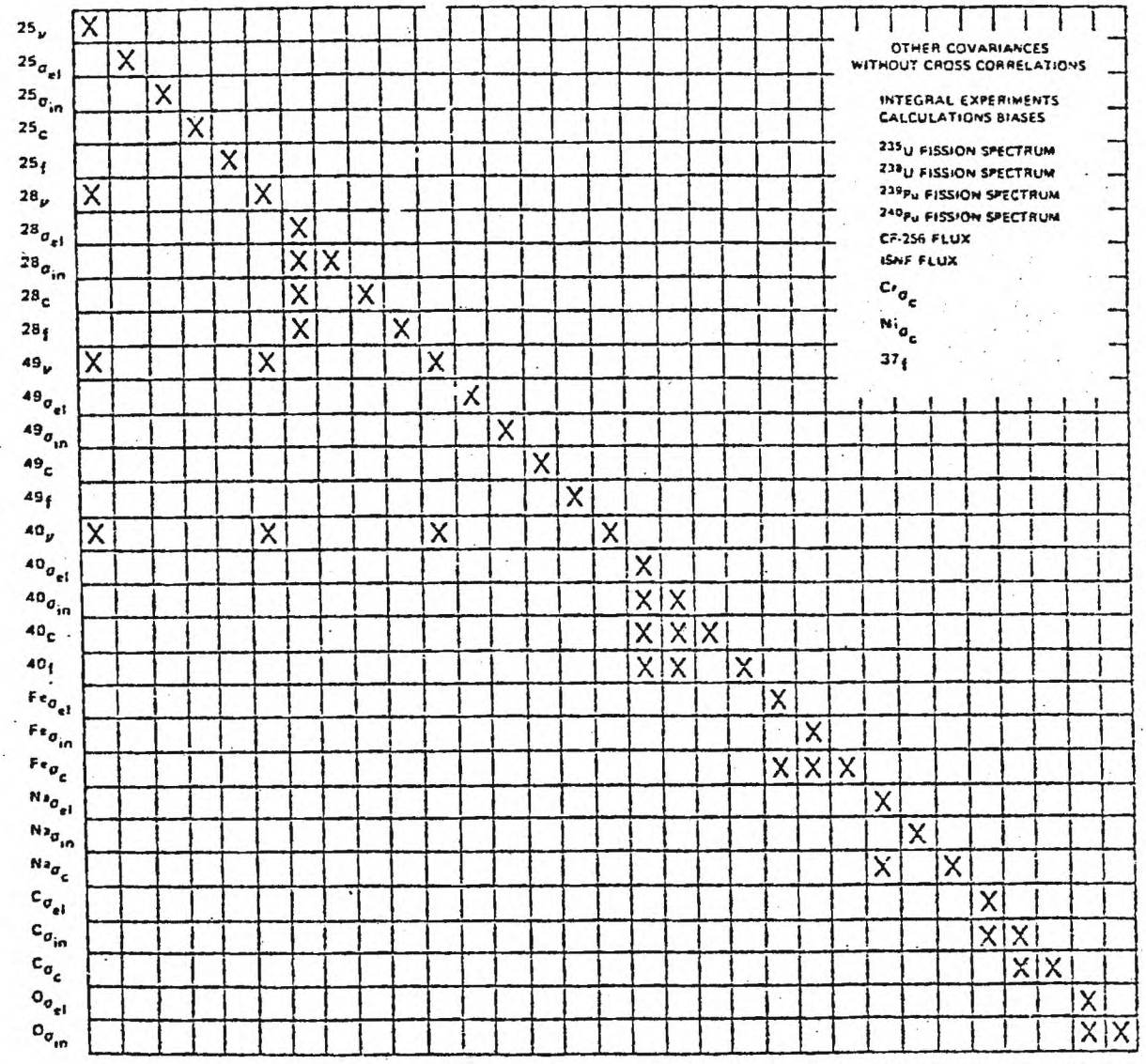
*See [M-1] and [K-1] for the reactions utilized for this cases.



[a]ABOVE AND TO THE RIGHT OF THE PRINCIPAL DIAGONAL X's SHOULD BE INFERRED FROM SYMMETRY. COLUMN LABELS ARE THE SAME AS ROW LABELS.

Fig. V-1. Covariance Data for Adjusted Library ORACLE-1

quite close to the corresponding reduction from 3.2% to 0.4% obtained for the heterogeneous LMFBR when using ORACLE-1. This indicates that inclusion of these thirteen integral experiments results in the adjustment procedure is responsible for the bulk of the resulting reduction in the standard deviation of k_{eff} for large fast reactors.

Uncertainty results due to nuclear data for all integral parameters considered in this study are shown in Table V-2. For comparison the previously reported uncertainties [K-2] based on unadjusted ENDF/B-IV data are also shown. As for the above discussion of k_{eff}, the reduction in the uncertainties of the breeding ratio is largely due to the effect of including the results of integral experiments. It must be emphasized that these uncertainties include only the effects of uncertainties in nuclear data as modified by the incorporation of integral experiments. Covariance contributions due to thermal-hydraulic and other engineering design uncertainties have not been included.

TABLE V-1

Standard Deviations for k_{eff}

Heterogeneous LMFBR		LCCEWG-LMFBR[b]	
ORACLE-I	ENDF/B-IV[a]	Unadjusted	Adjusted
0.4%	3.2%	3.1%	0.5%

a From Ref. K-2

b From Ref. M-1

TABLE V-2

Performance Parameter Uncertainties Based on ORACLE-1

	Standard Deviation, % (no k-reset)	
Parameter	ORACLE-1	ENDF/B-IV (Ref. K-2)
k_{eff}	0.4	3.2
BR	2	7
Sodium-void reactivity (all drivers)	14	---

SUMMARY AND CONCLUSIONS

The adjusted cross-section and covariance library ORACLE-I has been developed based on the Vitamin-E 174-group cross section library and an associated covariances. Some integral data which are not jointly consistent with the optimal response subset and the differential data base were discarded. This rejection is part of the "...intelligent, continuous intervention in the [adjustment] procedure..." necessary to avoid "...the pitfalls of 'blind' application of adjustment codes..." [S-1].

This adjusted library was applied to a sensitivity and uncertainty analysis of a large heterogeneous LMFBR. Responses included in this analysis were k_{eff}, the breeding ratio, and the sodium-void reactivity. A comparison of the uncertainty results for k_{eff} and the breeding ratios with previously reported uncertainties based on unadjusted data shows that the inclusion of integral experiments is significant for the reduction of uncertainties in calculated performance parameters due to nuclear data. This stresses the need for inclusion of integral experiments in design calculations.

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ORACLE REFERENCES

A-1. American National Standard ANS-19.3.4, (ANSI N676-1976), American Nuclear Society (1976).

B-1. L.A. Belblidia, J.M. Kallfelz and D.G. Cacuci, "Generalized Perturbation Theory with Derivative Operators for Power Density Investigations in Nuclear Reactors", Nucl. Sci. Engr. 84, 206(1983)..

B-2. G. B. Bruna, et al., Trans. ANS 24, 558 (1977).

B-3. A. T. D. Butland, W. N. Simmons and J. M. Stevenson, "An Assessment of Methods of Calculating Sodium-Voiding Reactivity in Plutonium-Fueled Fast Reactors", in Fast Reactor Physics, 1979", Vol. I, IAEA, Vienna (1980).

B-4. C. L. Beck, et al., "Sodium Voiding in LMFBR's: A Physics Assessment," ZPR-TM-400, Argonne National Laboratory, September 1, 1981.

B-5 J. Y. Barré, M. Heindler, T. Lacapelle, and J. Ravier, "Lessons Drawn from Integral Experiments on a Set of Multigroup Cross Sections," Proc. Conference on the Physics of Fast Reactor Operation and Design, British Nuclear Energy Society, London, June 23-26, 1969, Paper No. 1.15, pp. 165-179.

B-6 A. R. Baker, "Comparative Studies of the Criticality of Fast Critical Assemblies," Proc. of the International Conference on Fast Critical Experiments and Their Analysis, Argonne National Laboratory, October 10-13, 1966, ANL-7320, pp. 116-129.

C-1. D. G. Cacuci, C. F. Weber, E. M. Oblow, and J. H. Marable, "Sensitivity Theory for General Sytems of Nonlinear Equations", Nucl. Sci. Engr. 75., 88 (1980).

C-2. C. L. Cowan and C. S. Russell, "Fumble-II: A Fast Reactor Fuel Management and Burnup Code with an Influence Function Option for Predicting the Three-Dimensional Effects of Control Rod Movements", GEAP-14138, General Electric Co., Sunnyvale, CA, 1976.

C-3. G. P. Cecchini and M. Salvatores, Nucl. Sci. Eng. 46, 304 (1971).

C-4. D. G. Cacuci, "Sensitivity Theory of Nonlinear Systems, II. Extensions to Additional Classes of Responses," ORNL/TM-7511, Oak Ridge National Laboratory, August, 1981.

C-5. G. Cecchini, U. Farinelli, A. Gandini, and M. Salvatores, "Analysis of Integral Data for Few-Group Parameter Evaluation of Fast Reactors," Proc. of the Third United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1964, Vol. 2, Paper No. 627.

C-6. G. Cecchini, A. Gandini, I. Dal Bono, and B. Faleschini, "Comparison Between Experimental and Theoretical Integral Data on Fast Critical Facilities. CALI, A Program for Generating 'Effective' Nuclear Group Constants by a Correlation Method," Proc. of the International Conference on Fast Critical Experiments and Their Analysis, Argonne National Laboratory, October 10-13, 1966, ANL-7320, pp. 107-115.

C-7. C. G. Campbell, "Panel Discussion on Adjustments to Differential Data to Fit Integral Experiments," Proc. Conference on the Physics of Fast Reactor Operation and Design, British Nuclear Energy Society, London, June 23-26, 1969, pp. 277-280.

D-1. A. D'Angelo, L. Martin-Deidier, P. Marimbeau, J. P. Grouiller, E. Fort, R. Soule and M. Salvatores, "Calculation and Experiment Comparison for Sample and Fuel Pin Irradiation Experiments in Phenix", Proceedings of the Topical Meeting on Reactor Physics and Safety, Saratoga Springs, NY, Sept. 17-19, 1986, NUREG/CP-0080, U.S. Nuclear Regulatory Commission, 1986.

D-2. S. Davies, R. Murata, and B. Talwar, Trans. Am. Nucl. Soc. 34. (1980).

G-1. A. Gandini, G. Palmiotti and M. Salvatores, "Equivalent Generalized Perturbation Theory (EGPT". Ann. Nucl. Energy 13, 109 (1986).

G-2. H. Giese, C. Cavarec, G. Granget, and G. Palmiotti, "The Racine-1E Critical Experiments for Control-Rod Method and Data Validation Experimental and Calculation Results". Proceedings of the Topical Meeting on Reactor Physics and Safety, Saratoga Springs, NY, Sept. 17-19, 1986, NUREG/CP-0080, U.S. Nuclear Regulatory Commission, 1986.

G-3. A. Gandini, J. Nucl. Energy 21, 755 (1967).

G-4 A. Gandini and M. Salvatores, Nucl. Sci. Eng. 41, 452 (1970).

G-5. A. Gandini, et al., "Analysis of Fast Reactor by the CIAP and GLOBPERT Codes Using Improved Perturbation Methods," in ANL-7230, Argonne National Laboratory (1966).

G-6. A. Gandini, et al., "Sensitivity Study of Fast Reactors Using Generalized Pertubation Techniques," in Fast Reactor Physics, IAEA, Vienna, (1968).

H-1. H. H. Hummel and W. M. Stacey, Nucl. Sci. Engr.. 54, 35 (1974).

H-2. H. H. Hummel and D. Okrent, Reactivity Coefficients in Large Fast Power Reactors, American Nuclear Society, 1970. See Section 4-2.

H-3. P. C. E. Hemment and E. D. Pendlebury, "The Optimization of Neutron Cross Section Data Adjustments to Give Agreement with Experimental Critical Sizes," Proc. of the International Conference on Fast Critical Experiments and Their Analysis, Argonne National Laboratory, October 10-13, 1966, ANL-7320, pp. 88-106.

K-1. J. M. Kallfelz, M. L. Williams, D. Lal, and G. F. Flanagan, "Sensitivity Studies of the Breeding Ratio for the Clinch River Breeder Reactor", in J. M. Kallfelz and R. A. Karam, eds.; Advanced Reactors: Physics, Design and Economics, Pergamon Press, Oxford, New York, 1975.

K-2. J. M. Kallfelz, D. Biswas, C. L. Cowan, J. H. Marable, M. L. Williams, C. R. Weisbin, J. D. Drischler, T. B. Fowler, and J. R. White, "Design and Sensitivity Analysis of a CDS-Type Heterogeneous Core", in 1980 Advances in Reactor Physics and Shielding, American Nuclear Society, 1980.

K-3. J. M. Kallfelz, E. T. Tomlinson, D. Biswas, and J. R. White, "Preliminary Analysis and Sensivity Study of Phenix Fuel Irradiation Experiments", ORNL/TM-7505, ENDF-309, Oak Ridge National Laboratory, 1981.

K-4. "Depletion Analysis and Sensitivity Study of Phenix Fuel Irradiation Experiments", Trans. Am. Nucl. Soc. 33, 859 (1981).

K-5. E. Kujawski and R. Protsik, "A Generalized Treatment of Heterogeneity Effects for the Shielding Factor Method", Nucl. Tech. 41, 36 (1978).

K-6. H. Kusters and S. Ganesan, "Present Status of Sodium-Void Reactivity Predictions in Conventional and Nonconventional Fast Reactor Core Designs," in Advances in Reactor Physics, Proc. ANS Topical Meeting, Gatlinburg (1978).

K-7 E. Kujawski and H. S. Bailey, Nucl. Sci. Engr. 90 (1977).

M-1. J. H. Marable, D. G. Cacuci, C. R. Weisbin, J. R. White, D. Biswas, J. M. Kallfelz, C. L. Cowan and R. Protsik, "Sensitivities and Uncertainties of a Large LMFBR Using Adjusted Cross-Section Library ORACLE-1", Proc. Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics, NUREG/CP-0034, Vol II, U.S. Nuclear Regulatory Commission, p. 1120 (1982).

M-2. J. H. Marable, C. R. Weisbin and G. de Saussure, "Uncertainty in the Breeding Ratio of a Large Liquid-Metal Fast Breeder Reactor: Theory and Results", Nucl. Sci. Engr. 75, 30 (1980).

M-3. J. H. Marable, D. G. Cacuci, C. R. Weisbin, J. R. White, D. Biswas, J. M. Kallfelz, C. L. Cowan and R. Protsile, "Sensitivities and Uncertainties of a Large LMFBR Using Adjusted Cross-Section Library ORACLE-1", Trans. Am. Nucl. Soc. 41, 578 (1982).

M-4. J. H. Marable and C. R. Weisbin, "Uncertainties in the Breeding Ratio of a Large LMFBR", in E. G. Silver, ed., Advances in Reactor Physics, CONF-780401, U. S. Dept. of Energy, p. 231 (1978).

O-1. K. O. Ott, R. B. Pond and J. M. Kallfelz, "A Study of Methods of Cross-Section Error Identification Utilizing Integral Data from Fast Critical Assemblies" in Neutron Cross Sections and Technology (Proc. 3rd Conf. on this topic, Mar. 15-17, 1971, Knoxville, TN) CONF-710301, USAEC Division of Technical Information Extension, Oak Ridge, TN. Vol. I., p. 25.

O-2. K. O. Ott, R. B. Pond and J. M. Kallfelz, "A Study of Methods of Cross-Section Error Identification Utilizing Integral Data from Fast Critical Assemblies" in Neutron Cross Sections and Technology, ZPR-TM-87, Argonne National Laboratory, 1972.

P-1. R. W. Peele, "Report of the Working Group Specialist's Meeting on Methods and Practices in the Field of Nuclear Data Covariance", (Nov. 20, 1986 Draft), Private communication.

P-2. G. Palmiotti, and M. Salvatores, "Use of Covariance Methods and Practices in the Field of Nuclar Data, Proc. IAEA Specialists' Meeting on Covariance Methods and Practices in the Field of Nuclear Data, Rome, 17-19 November 1986.

P-3. R. Protsik, E. Kujawski and C. L. Cowan, "TDOWN-IV: A Computer Code to Generate Composition and Spatially Dependent Neutron Cross Sections for Multigroup Neutronics Analysis", GEFR-00485 (UC-78E), General Electric Co., 1979.

P-4. G. Palmiotti and M. Salvatores, "Use of Integral Experiments in the Assessment of Large Liquid-metal Fast Breeder Reactor Basic Design Parameters", Nucl. Sci Engr 87, 333 (1984).

P-5. W. P. Poenitz, "GMADJ, A Code for Generalized Least-squares Adjustments to Derived Quantities with Additional Experimental Data", ANL-ZPR-468, Argonne National Laboratory, April 1987.

R-1. G. Rimpault and G. Reynaud, "BD1: The Cadarache Data Bank for LMFBR Integral Experiment Data", (29th NEACRP Meeting, Chalk River, Sept 86).

R-2. J. L. Rowlands and J. D. Macdougall, "The Use of Integral Measurements to Adjust Cross Sections and to Predict Reactor Properties," Proc. Conference on the Physics of Fast Reactor Operation and Design, British Nuclear Energy Society, London, June 23-26, 1969, Paper No. 1.16, pp. 180-194.

R-3. J. L. Rowlands and H. Tubbs "The Joint Evaluated File: A New Nuclear Data Library for Reactor Calculations", Proc. Conf. on Nuclear Data for Basic and Applied Sciences, Santa Fe, NM, May 1985.

R-4. M. Robin et al., "Analyse des Combustibles Irradies du Reacteur Phenix", in Fast Reactor Physics 1979, Vol. I, International Atomic Energy Agency, Vienna, 1980.

S-1. U. Salmi, J. J. Wagschal, Atara Ya'ari and Y. Yeivin, "Controlled Cross-Section Adjustment by Integral Data", in "A Review of the Theory and Application of Sensitivity and Uncertainty Analysis", ORNL/RSIC-42, Oak Ridge National Laboratory, 1979, p. 49.

S-2. M. Salvatores, "Complementary Role of Critical Integral Experiment and Power Reactor Start-Up Experiments for LMFBR Neutronics Data and Method Validation", Proceedings of the Topical Meeting on Reactor Physics and Safety, Saratoga Springs, NY, Sept. 17-19, 1986, NUREG/CP-0080, U. S. Nuclear Regulatory Commission, 1986.

S-3. M. Salvatores and G. Palmiotti, LMFBR Design Parameter Uncertainties and Target Accuracies", Ann. Nucl. Energy 12, 291 (1985).

S-4. R. Soule, G. Granget, A. D'Angelo and H. Sztark, "The Experimental Balzac Program at Masurca in Support of the Design of Super Phenix 2", Proceedings of the Topical Meeting on Reactor Physics and Safety, Saratoga Springs, NY, Sept. 17-19, 1986, NUREG/CP-0080, U. S. Nuclear Regulatory Commission, 1986.

S-5. M. Salvatores, private communication, 23 Dec. 1986.

V-1. D. R. Vondy, T. B. Fowler, and G. W. Cunningham, "VENTURE," ORNL-5062/R1, Oak Ridge National Laboratory (1977).

W-1. C. R. Weisbin, R. W. Peele, J. H. Marable, P. Collins, E. Kujawski, E. Greenspan and G. deSaussure, "Sensitivity and Uncertainty Analysis of Reactor Performance Parameters", in J. Lewins and M. Becker, Eds. Advances in Nuclear Science and Technology, 14,

W-2. J. R. White, "The Development, Implementation, and Verification of Multicycle Depletion Perturbation Theory for Reactor Burnup Analysis," ORNL/TM-7305, Oak Ridge National Laboratory (1980).

Y-1. Y. Yeivin, J. H. Marable, C. R. Weisbin, and J. J. Wagschal, "ORACLE: An Adjusted Cross-Section and Covariance Library for Fast-Reactor Analysis", in 1980 Advances in Reactor Physics and Shiielding, American Nuclear Society, 1980, p. 467.

Y-2. S. Yiftah, "Panel Discussion on Adjustments to Differential Data to Fit Integral Experiments," Proc. Conference on the Physics of Fast Reactor Operation and Design, British Nuclear Energy Society, London, June 23-26, 1969, pp. 277-280.