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ARGONNE CODE CENTER:
COMPILATION OF PROGRAM ABSTRACTS

by

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This publication is a supplement to and revision of ANL-7411 issued in January 1968. Supplemental and replacement pages to be inserted in the original January publication are as follows:

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1-8	1-8	Revised cover, title page and table of contents
37-38, 38.1-38.2, 39-40	37-40	Reactor Code Abstract 33 revised
47-47.1, 47.2-48 76.1-76.2	47-48	Reactor Code Abstract 43 inserted Reactor Code Abstract 80 inserted
131-134	131-134	Reactor Code Abstract 143 revised
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195-198	195-198	Reactor Code Abstract 172 revised
199-202, 202.1- 202.2	199-202	Reactor Code Abstract 174 revised
297-298, 298.1- 298.2	297-298	Reactor Code Abstract 222 revised
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411-412	411-412	Reactor Code Abstract 278 revised
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<u>NO.</u>	<u>KWIC TITLE</u>	<u>INSTALLATION</u>	<u>MACH</u>	<u>LANG</u>	<u>PACKAGE</u>	
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9	FIRE2, 1-D AGE-DIFFUSION SLAB CYLINDER SPHERE	ROA	360	F4	RS PLT	24
14	HAFEVER, HAUSER-FESHBACH INELASTIC SCATTERING	CDC	1604	F63	RS P	26
18	2DXY, 2-D MULTI-GP SN APPROXIMATION XY GEOM	AGN	7090	FLOCO	RSBP	28
28	FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE	AI	7090	F2	RS P T	30
28	FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE	CDC	1604	F63	RS P T	30
29	AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	AI	7090	F+FAP	RS PLT	32
29	AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	CDC	1604	F63	RS PLT	32
30	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES	AI	7090	F2	RS	34
30	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES	CDC	1604	F63	RS	34
30	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES	BOEC	360	F4	RS P	34
32	WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY	ORNL	7090	F2	RS P T	36
32	WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY	CDC	1604	F63	RS P T	36
33	GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC	GA	7090	F2	RS PLT	38
33	GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC	CDC	1604	F63	RS PLT	38
33	GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC	ANL	3600	F63	RS LT	38
39	EQUIPOISE3, 2-D 2-GROUP DIFFUSION SLAB CYL	ORNL	7090	F2	RS P T	40
40	ZOGRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER	ORNL	7090	F2	RS P T	42
40	ZOGRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER	CDC	1604	F63	RS P T	42
41	ZUT, RESOLVED REGION RESONANCE INTEGRAL CALC	GA	7090	F+FAP	RSB T	44
42	TUZ, UNRESOLVED REGION RESONANCE INTEGRAL CALC	GA	7090	F+FAP	RSB T	46
R 43	CURFIT, CURVE FITTING EXPERIMENTAL DATA POINTS	KAPL	6600	F+ASC	RS P T	47.1
45	GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY	AI	7090	F2	RS P	48
45	GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY	CDC	1604	F63	RS P	48
46	GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM	AI	7090	F2	RS P	50
46	GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM	CDC	1604	F63	RS P	50
47	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	AI	7090	F2	RS P	52
47	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	CDC	1604	F63	RS P	52
47	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	SRL	360	F4	RS P	52
48	FUGUE, STEADY-STATE TEMPERATURE VOID FRACTION	AI	7090	F2	RS	54
50	TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS	AI	7090	F+FAP	RS PLT	56
50	TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS	CDC	1604	F63	RS PLT	56
50	TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS	BOEC	360	F4	RS PLT	56
51	FORM, FAST NEUTRON SPECTRUM X-SECTION CALC	AI	7090	F2	RS PLT	58
51	FORM, FAST NEUTRON SPECTRUM X-SECTION CALC	CDC	1604	F63	RS PLT	58
52	SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY	AI	7090	F2	RS	60
53	S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION	AI	7090	F2	RS	62
53	S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION	CDC	1604	F63	RS	62
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58	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	AI	7090	F2	RS LT		68
58	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	CDC	1604	F63	RS LT		68
58	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	AI	360	F4	RSBP T		68
59	MIST, 1-D FEW-GP SN DOUBLE SN APPROX SLAB GEOM	PP	7090	F2	RS P		70
62	LASL LEAST SQUARES, GENERAL CURVE FITTING	LASL	7094	F4	RS		72
75	GE-HAPO-S13, 1-D MULTI-GP DOUBLE SN APPROX	BNWL	7090	FLOCO	RSBP T		74
R 80	SOR2, STRESS ANALYSIS SHELLS OF REVOLUTION	KAPL	6600	F4	RS P T		76.1
87	EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER SLAB	ORNL	7090	F2	RS P T		77
87	EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER SLAB	CDC	1604	F63	RS P T		77
89	ARES2, RESONANCE INTEGRAL X-SECTION CALC	AI	7090	F2	RS PLT		79
89	ARES2, RESONANCE INTEGRAL X-SECTION CALC	CDC	1604	F63	RS PLT		79
91	FARSE1A, DOSE RATE FROM SNAP SHIELD LEAKAGE	AI	7090	F2	RS		81
99	DOB, 2-D FEW-GP DIFFUSION BURNUP RZ GEOMETRY	GA	7090	F+FAP	RSBP T		83
102	AX1, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE	CDC	3600	F63	RSBP		85
103	CRAM, 1-D AND 2-D MULTI-GP DIFFUSION PROGRAM	UK-R	7090	F+FAP	RSBP LT		87
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109	4RESTRAINT PIPE STRESS, MAXIMUM MOMENT CALC	AI	7090	F+FAP	RS T		92
110	SCARF2, SCATTER FROM RADIATOR FINS SNAP GEOM	AI	7090	F2	RS		93
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120	FAIMOS, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	BOEC	360	F4	RS P T		107
121	AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK	AI	7090	F2	RS		110
121	AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK	ROA	360	F4	RS P		110
122	SNAPKIN5/5A, 1-REGION KINETICS SNAP GEOMETRY	AI	7090	F2	RS		112
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135	TRAFICORPORATION, TRANSFER FUNCTION SYNTHESIS	ANL	3600	F63	RSBP T		124
136	HERESY1, LATTICE PARAMETERS HETEROGENEOUS CALC	FMA	7090	F2	RSBP		126
141	RATRAP, DOSE RATE CALCULATION SNAP GEOMETRY	AI	7090	F2	RS P		128
142	MORTIMER, DOSE RATE CALCULATION SNAP GEOMETRY	AI	7090	F2	RS		130
143	MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY	BNWL	7090	F2	RSBP LT		132
143	MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY	ANL	3600	F63	RSBP LT		132
144	DTF, 1-D MULTIGROUP DISCRETE ORDINATE CALC	UNC	1604	F63	RS PLT		135
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165	BURP2, DETECTOR EFFICIENCY DISK SOURCE	UM	7090	MAD	RSB	179
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167	FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION	GESJ	635	F4	RS P T	185
167	FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION	CDC	3600	F63	RSBP T	185
168	R102, SPACE-INDEPENDENT INVERSE KINETICS CALC	ANL	3600	F63	RSBP	187
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169	BURP4, GAMMA-RAY PHOTOFRACTION SOLID CRYSTAL	UM	7090	MAD	RSB	190
170	BURP5, GAMMA-RAY PHOTOFRACTION WELL CRYSTAL	UM	7090	MAD	RSB	192
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232	GAZELLE5, GAS-COOLED CORE THERMAL DESIGN STUDY	GA	7044	F4	RS P	317
233	CORE, CORE CONFIGURATION FUEL TEMPERATURE CODE	GA	7044	F4	RS	319
234	DPC, DATA PREPARATION FOR 2-D DESIGN PROGRAMS	LASL	7090	F2	RS P T	321
235	GAMMA-P, PRODUCTION X-SECTIONS FOR GAMMA-RAYS	GEC	7094	F4	RS P	323
236	DFSR, DATA FILE SERVICE ROUTINES ENDF TAPES	BNL	7090	F+FAP	RS P T	325
237	BOUNCE, FLUX DIST IN MULTI-PIN FUEL ELEMENT	AGN	7090	F2	RS P	327
238	EXT, X-SECTIONS FROM B-W RESONANCE PARAMETERS	WAL	7094	F2	RS P	329
239	CPS, SC4020 PLOTS FROM SCISRS X-SECTION TAPES	WAL	7094	F2	RS T	331
240	ASSAULT, 2-D MULTI-GP DIFFUSION DEPLETION CODE	ORNL	7090	F+FAP	RS P T	333
241	HFN, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	BNWL	1107	F4	RS P T	336
242	RATH, 2- OR 3-D HEAT CONDUCTION LUMPED MASS	LASL	7094	FAP	RS P T	338
243	AGN-SIGMA, CALC OF MULTI-GP TRANSFER MATRICES	AGN	7090	F2	RS P T	340
244	CYCLOPS1, THERMODYNAMIC CYCLE ANALYSIS	AI	7094	F2	RS PLT	342
246	FLOW-MODEL, MULTI-CHANNEL 2-D 2-PHASE FLOW	AI	7094	F2	RS P	344
247	FLANGE1, SCATTERING LAW X-SECTION CALCULATION	GA	7044	F4	RS	346
248	COINC, COINCIDENCE COUNTING DATA REDUCTION	ANL	3600	F63	RSBP	348
249	LASER, SPECTRUM CALC WITH BURNUP CYL LATTICE	WAPD	7094	F+MAP	RS PLT	350
250	SAFE-PCRS, STRESS ANALYSIS AXISYMMETRIC LOAD	GA	7044	F4	RS P T	353
251	SAFE-AXISYM, STRESS ANALYSIS AXISYMMETRIC LOAD	GA	7044	F4	RS P T	355
252	SAFE-PLANE, PLANE STRESS ANALYSIS, 2-D BODIES	GA	1108	F4	RS P	357
253	SAFE-SHELL, STRESS ANALYSIS THIN SHELLS	GA	1108	F4	RS P	359
254	2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS	AI	7094	F2	RS P T	361
254	2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS	ANL	3600	F63	RSBP T	361
255	R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS	ANL	3600	F63	RSBP	363
255	R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS	WAL	7094	F4	RS P	363
256	MANTA, STEADY-STATE THERMAL-HYDRAULIC ANALYSIS	GESJ	635	F4	RS P T	365
257	REAX, RESOLVED RESONANCE EPITHERMAL X-SECTIONS	GESJ	635	F4	RS P T	367
258	EXPN, ANALYSIS OF PULSED NEUTRON SOURCE DATA	GESJ	635	F4	RS P T	369
259	MUSCAT, VIEW FACTOR SHIELDING CODE CAVITY GEOM	GA	1108	F4	RS P T	371
260	GARGOYLE, FUEL CYCLE ANALYSIS PARTIAL REFUEL	GA	7044	F4	RS P T	373
261	GADOSE/DOSET, HTGR ACCIDENT ANALYSIS DOSE CALC	GA	1108	F4	RS T	375
262	MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	ANL	3600	F63	RS P T	377
263	GASKET, THERMAL SCATTERING LAW CALCULATION	GA	1108	F4	RS P	379
264	VARI-QUIR3, 2-D MULTI-GP DIFFUSION XY RZ RTH	WAL	7094	F4	RS P T	381
265	RSAC, RADIOLOGICAL SAFETY ANALYSIS PROGRAM	PP	7040	F+MAP	RS P T	383
R266	CYGR02, STRESS ANALYSIS CYL FUEL ELEMENT	BAPL	6600	F4	RS T	385
R267	WATER, STEAM TABLES 14.5-14,500PSIA 32-472DEGF	BAPL	2000	F4	RS	387
268	TRANS-FUGUE1, TRANSIENT FLOW AND HEAT TRANSFER	AI	7094	F2	RS P T	389
269	DTF-BURN, 1-D MULTI-GP DTF4 WITH DEPLETION	LASL	7030	F4	RS P T	391
270	CAESAR4/LIBLST, 1-D MULTI-GP DIFFUSION + LIB	AI	360	F4	RSBPLT	393
271	CLIP, FORM OR THREDES LIBRARY UTILITY ROUTINE	AI	360	F4	RSB LT	396
R272	FIGRO, LSBR FUEL SWELLING TEMPERATURE STUDY	BAPL	6600	F4	RS P T	398
273	THREDES, 1-D FEW-GP DIFFUSION DESIGN SYSTEM	AI	360	F4	RSBP T	400
R274	WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM	BAPL	6600	F4	RS P T	402
R275	PDQ7, 1,2 OR 3-D FEW-GP DIFFUSION DEPLETION	BAPL	6600	F+ASC	RS T	405
276	AVOID, ANNULAR VOID X-SECTION CALCULATION	GEC	625	F4	RS P	407

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277	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM	BNL-SRL	7090	F2	RS PLT	409
277	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM	BNL-SRL	360	F4	RS PLT	409
278	WHAM, LIQUID-FILLED PIPING SYSTEM ANALYSIS	KE	3600	F4	RSBP	411
278	WATER-HAMMER, LIQUID BLOWDOWN ANALYSIS	LOFT	UG	360	F4	RS P
279	LEOPARD, SPECTRA CALCULATION WITH DEPLETION	WAPD	7090	F+FAP	RSBPLT	413
R280	MO807, 2-D DIFFUSION ABSORPTION REMOVAL X-SECS	BAPL	6600	F4	RS T	415
281	RABBLE/WLIB/FLAT, RESONANCE ABSORPTION, CELL	ANL	3600	F36	RSBP T	417
R282	SEALSHLL2, SHELL STRESS ANALYSIS AXISYM LOAD	BAPL	6600	F4	RS P T	419
R283	MO552, DYNAMIC ANALYSIS LINEAR ELASTIC SYSTEMS	BAPL	6600	F4	RS P T	421
R284	MO555(ACT1), LOSS-OF-COOLANT ACCIDENT ANALYSIS	BAPL	6600	F4	RS P T	423
R285	RESQ2/RESQO/DBF1, RESONANCE INTEGRAL HEX CELL	BAPL	6600	F4	RS P T	425
R286	HOT2, 2-D TRANSIENT HEAT CONDUCTION PROGRAM	BAPL	6600	F4	RS P T	427
287	BISYN, 2-D MULTI-GP DIFFUSION SYNTHESIS CALC	GESJ	635	F4	RS PLT	429
288	SNARG-1D, 1-D MULTI-GP DISCRETE ORDINATE CALC	ANL	3600	F36	RS P T	432
289	GAKER, INELASTIC SCAT X-SECTION CALC MODERATOR	GGA	1108	F4	RS P	434
290	GASA, STABILITY ANALYSIS REACTOR KINETICS EQNS	GGA	1108	F4	RS P	436
291	HEXSCAT, ELASTIC SCAT X-SECTIONS HEX LATTICE	GGA	1108	F4	RS	438
292	PSEUDO, STATISTICAL RESONANCE PARAMETER CALC	GGA	1108	F4	RS P	440
293	MARS, 2-D EXCURSION CALCULATION R-Z GEOMETRY	APDA	7094	F4	RS P	442
R294	MO899/HOH, STEAM TABLES 14.5-2538 PSIA	BAPL	6600	F4	RS P T	444
R295	FLASH2, LOSS-OF-COOLANT ACCIDENT ANALYSIS	BAPL	6600	F4	RS P T	446
296	GRDWRK, GRID GENERATION FOR SAFE PROGRAMS	GGA	1108	F4	RS P	448
297	CONTEMPT, LOSS-OF-COOLANT ACCIDENT ANALYSIS	PPC	7040	F+MAP	RS P T	450
298	GGC4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA	GGA	1108	F4	RS PLT	452
R299	LION, 3-D TEMPERATURE DISTRIBUTION PROGRAM	KAPL	6600	F4	RS P T	456
300	SAFE-CREEP, VISCOELASTIC ANALYSIS CONCRETE	GGA	1108	F4	RS P	458
301	FREVAP6, HTGR METALLIC FISSION PRODUCT RELEASE	GGA	1108	F4	RS P	460
302	GAFFE, EQUILIBRIUM FUEL CYCLE CALCULATION	GGA	1108	F4	S T	462
303	BLOOST6, COMBINED KINETICS 2-D HEAT TRANSFER	GGA	1108	F4	RS P T	464
304	PERT4, 2-D PERTURBATION XY RZ RTHETA GEOMETRY	BNWL	1108	F4	RS P	466
305	STRIP, RESOLVED RESONANCE INTEGRAL CALCULATION	BW	2000	F4	RS	468
306	FCC4, FUNDAMENTAL MODE FAST REACTOR X-SEC CALC	BNWL	1108	F4	RS P T	470
307	HWOCR-SAFE, 2-D MONTE CARLO CELL CALCULATION	AI	360	F4	RS P T	472
308	JUPITOR1, COUPLED-CHANNEL X-SEC EVALUATION	ORNL	1604	F63	RS P T	474
309	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS	AI	7094	F4	RS P T	476
309	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS	AI	360	F4	RS P T	476
310	GAKIN, 1-D MULTIGROUP TIME-DEPENDENT DIFFUSION	GGA	1108	F4	RS P T	478
311	BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS	GEV	635	F4	RS P T	480
312	TDSN, 2-D MULTIGROUP DISCRETE ORDINATE PROGRAM	LRC	7090	F+MAP	RS P T	482
313	CINDER(MO102), POINT DEPLETION FISSION PRODUCT	BAPL	6600	F4	RS P	484
314	NAP, NEUTRON-INDUCED GAMMA-RAY RADIOACTIVITY	IITR	7094	F4	RSBPLT	486
315	WIREX, COMPUTER-PRODUCED WIRING LISTS UHTREX	LASL	7090	F2	RS P	488
316	GAF/GAR, SPECTRA AND GROUP-AVERAGED X-SEC CALC	GGA	1108	F4	RS P T	490
317	GAPOTKIN, SPACE-INDEPENDENT REACTOR KINETICS	GGA	1108	F4	RS P	493
318	FEVER7, 1-D MULTIGROUP DIFFUSION AND DEPLETION	GGA	1108	F4	RS P T	495
319	GASP7, 1-D BURNUP POWER DISTRIBUTION SEARCH	GGA	1108	F4	RS P T	497
320	TEMCO7, TEMPERATURE COEFFICIENT CALCULATION	GGA	1108	F4	RS P T	499
321	EXPALs, LEAST SQUARES EXPONENTIAL DECAY CURVES	LRL	7094	F2	RS P	501
322	ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC	ORNL	1604	F63	RS P	503
322	ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC	ORNL	360	F4	RS P	503
323	MUFFLE, FISSION NUCLIDE X-SECTION EVALUATION	ORNL	7090	F4	RSBP	505
324	FRANTIC, LEAST SQUARES FIT SUM OF EXPONENTIALS	MIT	7090	F2	RS P	507
325	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	BNWL	1108	F4	RS P T	509
326	AIROS2A, SIMULATION OF REACTOR DYNAMICS	AI	360	F+BAL	RS P T	511
R327	DAFT1, LEAST SQUARES FIT FISSION NUCLIDE DATA	BAPL	6600	F4	RS P T	514
328	MURLOC-1.0, LOSS-OF-COOLANT THERMAL ANALYSIS	BMI	6400	F4	RS P T	516
R329	MO457(PIPE), ELASTIC STRESS OF PIPING SYSTEM	BAPL	6600	F4	RS P T	518
330	ECCSA1, LOSS-OF-COOLANT + EMERGENCY COOLING	BMI	6400	F4	RS P T	520
R331	FLOT1(MO219), PWR FLOW TRANSIENT ANALYSIS	BAPL	6600	F4	RS P T	522
332	SAFE-3D, 3-D COMPOSITE STRUCTURE STRESS STUDY	GGA	1108	F4	RS P T	524
333	TAD, PROCESSING OF ANALYZER GAMMA-RAY SPECTRA	GGA	1108	F4	RS P T	526
334	PEGGY, ELASTIC SCATTERING PHASE-SHIFT ANALYSIS	ORNL	1604	F63	RS P	528
335	RAMES, PARTICLE WAVE FUNCTION RADIAL INTEGRALS	ORNL	1604	F63	RS P	530
R336	PDQ5, 2-D FEW-GROUP DIFFUSION AND DEPLETION	IBM	360	F+BAL	RS P T	531

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15. NAME AND ESTABLISHMENT OF AUTHORS (CONTINUED)

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECKS (7090-2771 CARDS AND 1604-2809 CARDS)

SAMPLE PROBLEMS (7090-11 CARDS AND 1604-11 CARDS)

REFERENCE REPORT, ERRATA, ADDENDUM, AND NOTES

17. CATEGORY - C

KEYWORDS - DIFFUSION EQUATIONS, 3-DIMENSIONAL, 2-GROUP, X-Y-Z,
FLUX DISTRIBUTION, REACTIVITY

1. NAME OR DESIGNATION OF PROGRAM - GAM1/REP/UPDATE
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090, CDC1604,3600
3. NATURE OF PHYSICAL PROBLEM SOLVED - THIS PROGRAM COMPUTES THE SLOWING-DOWN SPECTRUM IN EITHER THE P1 OR THE B1 APPROXIMATION USING 68 GROUPS OF NEUTRONS WITH A CONSTANT GROUP WIDTH $DU=0.25$, MULTIGROUP CONSTANTS ARE CALCULATED FOR UP TO 32 FAST GROUPS.
4. METHOD OF SOLUTION - SPECIAL FEATURES OF THIS CODE INCLUDE -
 - (A) A SUBROUTINE THAT CALCULATES RESONANCE INTEGRALS FROM THE RESONANCE PARAMETERS FOR EACH GROUP USING THE QUANTITATIVE METHODS DEVELOPED BY L. W. NORDHEIM.
 - (B) THE ENERGY ANGLE CORRELATION IS RETAINED FOR SLOWING-DOWN IN ALL ISOTOPES. THAT IS, THE P0 AND P1 COMPONENTS OF THE SCATTERING KERNEL ARE TREATED RIGOROUSLY. FOR THE COMMON MODERATING MATERIALS AND FOR SOME OF THE MORE COMMON ISOTOPES, TERMS UP TO P6 IN THE LEGENDRE EXPANSION OF THE SCATTERING CROSS SECTION IN THE CENTER OF MASS SYSTEM ARE RETAINED. FOR OTHER ISOTOPES, THE SCATTERING MATRICES ARE CORRECT FOR LINEARLY ANISOTROPIC SCATTERING IN THE CENTER OF MASS SYSTEM.
 - (C) THE CODE CONTAINS AN OPTION THAT WILL CALCULATE THE NEUTRON AGE IN AN INFINITE MEDIUM BY THE MOMENTS METHOD.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - NINE DIFFERENT SOURCE SPECTRA ARE INCLUDED AS OPTIONS.
6. TYPICAL RUNNING TIME - 1 TO 5 MINUTES ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM - THE CODE CONTAINS A LARGE LIBRARY. AT PRESENT, THE LIBRARY TAPE CONSISTS OF 127 NUCLIDES. THE LIBRARY TAPE CONTAINS ALL OF THE DATA NEEDED IN THE CALCULATION FOR EACH OF THE 68 SUBGROUPS. INELASTIC SCATTERING AND (N,2N) PROCESSES ARE EXPLICITLY INCLUDED. SELF-SHIELDING FACTORS MAY BE INCLUDED FOR ANY NUCLIDE, IF DESIRED.
8. RELATED AND AUXILIARY PROGRAMS - REP, TO REPRODUCE GAM LIBRARY TAPE FOR THE 7090 AND 1604, UPDATE, TO UPDATE GAM LIBRARY TAPE FOR THE 7090, AND GAMLIBM, TO PREPARE, PRINT, AND EDIT GAM LIBRARY TAPES FOR THE 3600.
9. STATUS - ABSTRACT FIRST DISTRIBUTED FEBRUARY 1962.
IBM7090 VERSION SUBMITTED FEBRUARY 1962.
CDC1604 VERSION SUBMITTED AUGUST 1963.
CDC3600 VERSION SUBMITTED FEBRUARY 1968.
10. REFERENCES - G. D. JOANOU, E. J. LESHAN, AND J. S. DUDEK, GAM-1, A CONSISTENT P1 MULTIGROUP CODE FOR THE CALCULATION OF THE FAST NEUTRON SPECTRUM AND MULTIGROUP CONSTANTS, GA-1850, MAY 1961.

10. REFERENCES (CONTINUED)

- G. D. JOANOU, J. S. DUDEK, A. J. GOODJOHN,
N. F. WIKNER, AND J. D. GARRISON, NUCLEAR DATA FOR GAM-1 DATA
TAPE, GA-2451, VOLUMES I AND II, AUGUST 1961.
- F. T. ADLER, G. W. HINMAN, AND L. W. NORDHEIM, THE
QUANTITATIVE EVALUATION OF RESONANCE INTEGRALS, PROCEEDINGS OF THE
SECOND UNITED NATIONS INTERNATIONAL CONFERENCE ON PEACEFUL USES OF
ATOMIC ENERGY, VOL. 16, PP. 155-170, GENEVA, 1958.
- FORMAT OF GAM-1 BCD LIBRARY INFORMATION, ARGONNE CODE
CENTER NOTE, AUGUST 10, 1966.
- GAM-1, NOTES CONCERNING THE CONTROL DATA VERSION.
- G. JENSEN, GAM-1 (R104) USAGE ON 3600, ANL NOTE,
DECEMBER 2, 1964.
- G. JENSEN, R104 GAM LIBRARY PREPARATION, ANL NOTE,
APRIL 12, 1965.

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGES USED - FORTRAN II (IBM7090), FORTRAN 60 AND
FORTRAN 63 (CDC1604 AND CDC3600)

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
FORTRAN MONITOR SYSTEM (IBM7090), COOP MONITOR (CDC1604), SCOPE
(CDC3600).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (7090-819 CARDS, 1604-1478 CARDS, 3600-1575 CARDS)
SAMPLE PROBLEMS (7090-21 CARDS, 1604-18 CARDS)
LIBRARIES (BCD 7090 AND 1604-16607 CARDS EACH, BINARY 3600-797
BINARY RECORDS)
REP SOURCE DECKS (7090-223 CARDS, 1604-46 CARDS)
UPDATE SOURCE DECK (7090-588 CARDS)
GAMLIBM SOURCE DECK (3600-329 CARDS)
REFERENCE REPORT, GA-1850, AND NOTES
MEMORANDA - THE STRUCTURE OF THE GAM DATA TAPE,
GAM CODE,
GAM LIBRARY TAPE,
NEW GAM INPUT.
17. CATEGORY - 8
KEYWORDS - SLOWING DOWN SPECTRA, FAST NEUTRONS, MULTIGROUP GROUP
CONSTANTS, RESONANCE INTEGRALS, ANISOTROPIC SCATTERING,
AGE, PL METHOD, BL METHOD, REP CODES, UPDATE CODES

1. NAME OR DESIGNATION OF PROGRAM - EQUIPOISE3
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090.
3. NATURE OF PHYSICAL PROBLEM SOLVED - EQUIPOISE3 SOLVES THE TWO-GROUP, TWO-DIMENSIONAL, NEUTRON DIFFUSION EQUATIONS IN CYLINDRICAL OR SLAB GEOMETRY.
4. METHOD OF SOLUTION - THE EQUIPOISE METHOD DESCRIBED IN REFERENCE 2 BELOW IS USED TO SOLVE THE FINITE-DIFFERENCE ANALOGS OF THE TWO-GROUP NEUTRON DIFFUSION EQUATIONS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
 - MAXIMUM NUMBER OF GROUPS = 2
 - MAXIMUM NUMBER OF MESH POINTS = 2100
 - MAXIMUM NUMBER OF COMPOSITIONS = 50
 - MAXIMUM NUMBER OF REGIONS = 100
6. TYPICAL RUNNING TIME - APPROXIMATELY 0.0017 SECOND PER POINT PER ITERATION IS REQUIRED, ABOUT 5 MINUTES FOR A 1000-POINT PROBLEM.
7. UNUSUAL FEATURES OF THE PROGRAM - THIS CODE IS INTENDED TO FILL THE NEED FOR A RAPID TWO-DIMENSIONAL CALCULATION SUITABLE FOR SURVEY CALCULATIONS. MAGNETIC TAPES ARE USED ONLY FOR INPUT, OUTPUT, AND PROGRAM STORAGE. THE LOGARITHMIC DERIVATIVE BOUNDARY CONDITION IS ALLOWED, AND ADJOINT FLUXES AND ASSOCIATED DIRECT-ADJOINT FLUX REGION INTEGRALS MAY BE CALCULATED AUTOMATICALLY IF DESIRED.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED APRIL 1962.
IBM7090 VERSION SUBMITTED FEBRUARY 1962.
10. REFERENCES - T. B. FOWLER AND M. L. TOBIAS, EQUIPOISE-3, A TWO-DIMENSIONAL, TWO-GROUP NEUTRON DIFFUSION CODE FOR THE IBM-7090 COMPUTER, ORNL-3199, FEBRUARY 1962.
M. L. TOBIAS AND T. B. FOWLER, THE EQUIPOISE METHOD - A SIMPLE PROCEDURE FOR GROUP-DIFFUSION CALCULATIONS IN TWO AND THREE DIMENSIONS, NUCLEAR SCIENCE AND ENGINEERING, VOL. 12, PP. 513-518, 1962.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN II (7090)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - IBM FORTRAN II MONITOR (7090).

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17. CATEGORY - A

KEYWORDS - UNRESOLVED REGION RESONANCE INTEGRALS, TEMPERATURE

1. NAME OR DESIGNATION OF PROGRAM - CURFIT
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - CURFIT IS A COMPOSITE PROGRAM FOR FITTING EXPERIMENTAL DATA POINTS WITH DIFFERENT TYPES OF COMMON ANALYTIC CURVES. THERE ARE AT PRESENT FIVE FITS AVAILABLE -
 - (1) POLYNOMIAL $Y = \text{SUMMATION OVER } I \text{ OF } A(I) \cdot (X^{**}(I))$
 - (2) EXPONENTIAL $Y = A \cdot \text{EXP}(BX)$
 - (3) COSINE $Y = A \cdot \text{COS}(B(X+C))$
 - (4) SERIES OF CUBICS $Y = A(J) + B(J) \cdot X + C(J) \cdot (X^{**}2) + D(J) \cdot (X^{**}3)$
 - (5) FOURIER SERIES
4. METHOD OF SOLUTION - THE SOLUTION VARIES WITH THE FIT. CURFIT PRODUCES POLYNOMIAL COEFFICIENTS BY USING ORTHOGONAL POLYNOMIALS. THE SINGLE EXPONENTIAL AND THE COSINE FITS EMPLOY THE LEAST SQUARES METHOD. IN THE SERIES OF CUBIC FITS THE CURVE IS A COMPOSITE OF CUBIC PARABOLAS HAVING CONTINUOUS FIRST AND SECOND DERIVATIVES THROUGHOUT WITH THE SECOND DERIVATIVES AT THE END POINTS SET TO ZERO. THE FOURIER SERIES METHOD USED WAS ONE DEVELOPED BY J. P. JEWETT.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
6. TYPICAL RUNNING TIME - LESS THAN 15 SECONDS PER CASE ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - A PREVIOUS VERSION EXISTED FOR THE PHILCO 2000 COMPUTER.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC6600 VERSION SUBMITTED JUNE 1968.
10. REFERENCES - J. A. WARRINGTON, CURFIT, CURVE FITTING PROGRAM, KAPL-M-6483 (EC-55), JULY 15, 1965.
R. J. CULLEN, 6600 CALCOMP PLOTTER ROUTINES, KAPL NOTE, JULY 25, 1966.
J. A. WARRINGTON, CONVIX, A FORTRAN 4 FORMAT FREE, SYMBOLICALLY ADDRESSED INPUT ROUTINE, KAPL/DAC/CT INFORMAL PROGRAM DESCRIPTION, JUNE 17, 1968.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASCENT
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 2.C.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - INPUT PREPARATION IS EASY, DUE TO USE OF A COMMA LOADER.

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15. NAME AND ESTABLISHMENT OF AUTHOR -

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GENERAL ELECTRIC COMPANY
SCHENECTADY, NEW YORK

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION

MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (3621 CARDS)
SAMPLE PROBLEM (69 CARDS)
REFERENCE REPORT, NOTE, PROGRAM DESCRIPTION

17. CATEGORY - P

KEYWORDS - LEAST SQUARES, DECAY, BUCKLING, DATA PROCESSING

1. NAME OR DESIGNATION OF PROGRAM - GRACE1
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090, CDC1604
3. NATURE OF PHYSICAL PROBLEM SOLVED - GRACE1 IS A MULTIGROUP, MULTI-REGION, GAMMA-RAY ATTENUATION PROGRAM DESIGNED PRIMARILY FOR COMPUTING GAMMA-RAY HEATING AND GAMMA-RAY DOSE RATES IN MULTIREGION FINITE OR SEMI-INFINITE SLAB SHIELDS. A DIFFERENT BUILDUP FACTOR MAY BE SPECIFIED FOR EACH SOURCE REGION CONSIDERED.
4. METHOD OF SOLUTION -
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
 - MAXIMUM NUMBER OF REGIONS = 30
 - MAXIMUM NUMBER OF MESH POINTS IN EACH REGION = 10
 - MAXIMUM NUMBER OF GAMMA-RAY ENERGY GROUPS = 20
 - MAXIMUM NUMBER OF SHIELD MATERIALS = 10
 - MAXIMUM NUMBER OF MATERIAL BUILDUP FACTORS = 5
6. TYPICAL RUNNING TIME - A SAMPLE PROBLEM INVOLVING 1 SOURCE REGION, 9 MESH POINTS AND 1 ENERGY GROUP REQUIRED 39 SECONDS ON THE 7090.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - GRACE2 (ACC ABSTRACT 46)
9. STATUS - ABSTRACT FIRST DISTRIBUTED APRIL 1962.
 - IBM7090 VERSION SUBMITTED MARCH 1962.
 - CDC1604 VERSION SUBMITTED MARCH 1964.
10. REFERENCES - D. S. DUNCAN AND A. B. SPEIR, GRACE I, AN IBM 704-709 PROGRAM DESIGNED FOR COMPUTING GAMMA-RAY ATTENUATION AND HEATING IN REACTOR SHIELDS, NAA-SR-3719, 1959.
 - J. AGRESTA, GRACE-I NOTE, UNION CARBIDE CORPORATION.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGES USED - FORTRAN II (IBM7090) AND FORTRAN 63 (CDC1604)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
 - 7090
 - R. A. BLAINE
 - SUPERVISOR, REACTOR COMPUTING SYSTEMS UNIT
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 - ATOMICS INTERNATIONAL
 - P. O. BOX 309
 - CANOGA PARK, CALIFORNIA

REACTOR CODE ABSTRACT 80

6/68

1. NAME OR DESIGNATION OF PROGRAM - SOR2
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - SOR2 SOLVES FOR THE FORCES, STRESSES, DEFLECTIONS, AND STRAINS IN THIN SHELLS OF REVOLUTION. THE SHELLS MAY BE GENERAL SURFACES OF REVOLUTION WITH VARIABLE THICKNESSES AND ELASTIC MODULI. THIS INCLUDES THE MORE FAMILIAR FORMS - THE CIRCULAR FLAT PLATE, CONE, CYLINDER, SPHERE, ELLIPSE, AND TORII WITH CIRCULAR OR ELLIPTICAL CROSS SECTIONS, FOR WHICH A SIMPLIFIED INPUT IS USED. THE AXISYMMETRIC LOADINGS CONSIDERED INCLUDE ARBITRARY DISTRIBUTIONS OF NORMAL, TANGENTIAL AND MOMENT SURFACE LOADINGS, AS WELL AS EDGE FORCES AND DEFLECTIONS. THE EFFECTS OF RADIAL AND AXIAL TEMPERATURE DISTRIBUTIONS, CENTRIFUGAL LOADING DUE TO ROTATION ABOUT THE AXIS AND VIBRATION ARE INCLUDED. THE ADDITIONAL EFFECTS OF MISALIGNMENT, LINE LOADS, AND ELASTIC SUPPORTS AT THE SHELL INTERSECTIONS ARE CONSIDERED.
4. METHOD OF SOLUTION - THE PROGRAM NUMERICALLY INTEGRATES E. REISSNER'S EQUATIONS USING A GENERALIZED ADAMS-MOULTON METHOD (AMSINT SUBROUTINE). THE INTEGRATION PROCEEDS ALONG THE SHELL WITH AUTOMATIC ADJUSTMENT OF INTERVAL SIZE FOR MAXIMUM SPEED WITHIN THE LIMITATIONS OF THE PRESET ERROR BOUNDS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - SOR2 WILL SOLVE AND JOIN TOGETHER UP TO 30 SHELLS. THE INTEGRATION ERROR BOUNDS, WHICH MAY BE ALTERED BY INPUT, ARE PRESET AT 5 SIGNIFICANT FIGURES. TO INSURE OPTIMUM RESULTS, IT IS RECOMMENDED THAT THE MAXIMUM LENGTH OF ANY ONE SHELL NOT EXCEED 6 ATTENUATION LENGTHS.
6. TYPICAL RUNNING TIME - AN AVERAGE TIME OF 10 SECONDS PER CASE IS REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM - THE METHOD USED PERMITS SOLUTION OF PROBLEMS IN WHICH THE SHELL INCLUDES THE AXIS OF REVOLUTION. IT ALSO ALLOWS STEP CHANGES IN LOADINGS AND SOME SHELL PROPERTIES.
8. RELATED AND AUXILIARY PROGRAMS - A PREVIOUS VERSION OF THE SOR2 PROGRAM EXISTED FOR THE PHILCO 2000.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC6600 VERSION SUBMITTED JUNE 1968.
10. REFERENCES - J. A. MIRABAL AND D. G. DIGHT, SOR-II - A PROGRAM TO PERFORM STRESS ANALYSIS OF SHELLS OF REVOLUTION, KAPL-2292, SEPTEMBER 15, 1963.
R. L. FAGAN, SOR-II INPUT AND MODIFICATIONS, KAPL NOTE, FEBRUARY 9, 1968.
11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
SCOPE 2.0.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (1640 CARDS)
SAMPLE PROBLEM (81 CARDS)
REFERENCE REPORT AND NOTE
17. CATEGORY - I
KEYWORDS - STRESSES, SHELLS, SPHERES, CYLINDERS, TEMPERATURE DIS-
TRIBUTION

REACTOR CODE ABSTRACT 142

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10/67

15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE -

SOURCE DECK (1209 CARDS)
REFERENCE REPORT

17. CATEGORY - J

KEYWORDS - DOSE RATES, SNAP REACTORS, SHIELDING

1. NAME OR DESIGNATION OF PROGRAM - MAC
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090, CDC3600
3. NATURE OF PHYSICAL PROBLEM SOLVED - MAC PERFORMS SLAB GEOMETRY, MULTIGROUP NEUTRON AND GAMMA RAY PENETRATION ANALYSIS FOR A MULTI-REGION REACTOR SHIELD. THE CODE CALCULATES THE FOLLOWING INFORMATION AS A FUNCTION OF DISTANCE THROUGH A REACTOR SHIELD ASSEMBLY -
NEUTRON FLUXES FOR UP TO 35 ENERGY GROUPS,
NEUTRON DOSE RATES,
THE APPROXIMATE NEUTRON SPECTRUM,
GAMMA RAY FLUXES FOR 7 ENERGY GROUPS,
TOTAL GAMMA DOSE RATE, WITH A BREAKDOWN OF THE CONTRIBUTION FROM SOURCES IN EACH REACTOR AND SHIELD REGION,
THE APPROXIMATE GAMMA RAY SPECTRUM.
4. METHOD OF SOLUTION - MAC IS WRITTEN TO CALCULATE MULTIGROUP NEUTRON FLUX BY A COMBINED REMOVAL THEORY AND A MULTIGROUP DIFFUSION ROUTINE WHERE THE FISSION NEUTRON SPECTRUM IS BROKEN INTO 18 ENERGY GROUPS AND THE SPATIAL DISTRIBUTION THROUGH THE SHIELD REGIONS OF THE FIRST NEUTRON COLLISION IS CALCULATED WITH A REMOVAL KERNEL AND ENERGY DEPENDENT COLLIMATED REMOVAL CROSS SECTIONS. FIRST COLLISION NEUTRONS ARE THEN PLACED IN A PROPER DIFFUSION ENERGY GROUP AND ISOTROPIC FLUX DISTRIBUTIONS AND NEUTRON CAPTURE DISTRIBUTIONS ARE CALCULATED. INELASTIC SCATTERING IS ACCOUNTED FOR WITH THE STATISTICAL EVAPORATION MODEL OF WEISSKOPF. MAC CALCULATES THE PRODUCTION OF GAMMA RADIATION IN THE REACTOR AND SHIELDS ARISING FROM THE FISSION PROCESS AND NEUTRON CAPTURE PROCESS. FOR CAPTURE GAMMA RAY PRODUCTION, THE CODE CALCULATES THE CAPTURE GAMMA RAY SOURCE AS A FUNCTION OF SPACE AND ENERGY DISTRIBUTION BY APPLYING THE NUMBER OF GAMMA RAYS PRODUCED IN EACH OF 7 GAMMA RAY ENERGY GROUPS FOR EACH NEUTRON CAPTURE IN EACH TYPE OF NUCLIDE. ALL GAMMA RAY SOURCES ARE THEN ATTENUATED THROUGH THE SHIELD LAYERS USING STANDARD ATTENUATION EQUATIONS WITH TAYLOR BUILDUP FACTORS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
MAXIMUM NUMBER OF SHIELD REGIONS = 20
MAXIMUM NUMBER OF NEUTRON ENERGY GROUPS = 35
MAXIMUM NUMBER OF ELEMENTS = 21
MAXIMUM NUMBER OF GAMMA RAY ENERGY GROUPS = 7
MAXIMUM NUMBER OF INELASTIC SCATTERING POSSIBILITIES FROM EACH OF THE TOP NEUTRON ENERGY GROUPS = 6
MAXIMUM NUMBER OF MESH POINTS = 2200
6. TYPICAL RUNNING TIME - AVERAGE PROBLEMS MAY BE SOLVED IN FROM 5 TO 20 MINUTES. SAMPLE PROBLEM EXECUTION REQUIRES 16 MINUTES.
7. UNUSUAL FEATURES OF THE PROGRAM - THE ENTIRE REACTOR SHIELD PROBLEM IS SOLVED IN ONE COMPUTER CALCULATION. TWO CROSS SECTION LIBRARIES ARE AVAILABLE CONTAINING COMPLETE CROSS SECTION AND GROUP

7. UNUSUAL FEATURES OF THE PROGRAM (CONTINUED)
CONSTANT DATA FOR 77 ELEMENTS AND 7 ISOTOPES. THE ABSORPTION, ELASTIC SCATTERING, AND INELASTIC SCATTERING NEUTRON CROSS SECTIONS ARE GIVEN IN 15 AND 31 GROUP ENERGY-LETHARGY STRUCTURES. THE ADDITIONAL GROUP CONSTANTS ARE COMMON TO BOTH LIBRARIES AND INCLUDE - GAMMA RAY MASS ATTENUATION COEFFICIENTS (7 GROUP), ENERGY DEPENDENT REMOVAL CROSS SECTIONS (18 GROUP), NEUTRON AND GAMMA RAY FLUX TO DOSE CONVERSION FACTORS, TAYLOR BUILDUP FACTORS, AND THE GAMMA RAYS PRODUCED PER NEUTRON CAPTURE PER NUCLIDE.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED NOVEMBER 1964.
IBM7090 VERSION SUBMITTED MARCH 1964.
CDC3600 VERSION SUBMITTED JANUARY 1968, SAMPLE PROBLEM EXECUTED BY ACC.
10. REFERENCES - R. G. PETERSON, MAC - A BULK SHIELDING CODE, HW-73381, APRIL, 1962.
J. GREENBORG, TWO CROSS SECTION LIBRARIES FOR USE WITH MAC SHIELDING CODE, HW-73381, SUPPLEMENT, JANUARY, 1964.
J. GREENBORG, ADDITIONS AND ERRATUM TO HW-73381, SUPPLEMENT 1, NOVEMBER 13, 1964.
H. YAMANOUCHI AND A. E. MCARTHY, RP-370X, ANL REACTOR PHYSICS MEMO, AUGUST 27, 1965.
H. YAMANOUCHI, RP-370X ADDITION, ANL REACTOR PHYSICS MEMO ADDITION, SEPTEMBER 10, 1965.
11. MACHINE REQUIREMENTS - 32K MEMORY AND 3 TAPE UNITS (INPUT, OUTPUT, AND SCRATCH)
12. PROGRAMMING LANGUAGES USED - FORTRAN II (IBM7090) AND FORTRAN IV (CDC3600)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - STANDARD IBM FORTRAN II MONITOR (IBM7090) AND SCOPE (CDC3600).
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
CODE E. G. PETERSON
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECKS (7090-1167 CARDS, 3600-1305 CARDS)
BINARY DECKS (7090-291 CARDS, 3600-528 CARDS)
SAMPLE PROBLEMS (7090-617 CARDS, 3600-616 CARDS)
LIBRARIES (7090-3214 CARDS, 3600-3221 CARDS)
REFERENCE REPORTS AND MEMORANDA

17. CATEGORY - J

KEYWORDS - DIFFUSION EQUATIONS, 1-DIMENSIONAL, MULTIGROUP, SLABS,
GAMMA RADIATION, NEUTRONS, SPECTRA, SHIELDING, DOSE
RATES

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12. PROGRAMMING LANGUAGES USED - FORTRAN II (95 PER CENT)
FAP (5 PER CENT)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
STANDARD IBM FORTRAN MONITOR.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
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P. O. BOX 309
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
GRAVE SOURCE DECK, INCLUDING ARES II (3500 CARDS)
SAMPLE PROBLEM (23 CARDS)
ARES LIBRARY (500 CARDS)
REFERENCE REPORTS, NAA-SR-MEMO-9276 AND NAA-SR-MEMO-8889, AND
LETTER
17. CATEGORY - B
KEYWORDS - CROSS SECTIONS AVERAGES, MAXWELL DISTRIBUTION, MULTI-
GROUP GROUP CONSTANTS

1. NAME OR DESIGNATION OF PROGRAM - AIROS
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7094
3. NATURE OF PHYSICAL PROBLEM SOLVED - AIROS SOLVES THE SPACE-INDEPENDENT REACTOR KINETICS EQUATIONS AND PROVIDES FOR THE DETERMINATION OF REACTIVITY BY SOLVING IN ADDITION THE DISCRETIZED EQUATIONS WHICH REPRESENT THE SPATIAL HEAT AND MASS TRANSFER MODEL FOR SEVERAL FUEL CHANNELS. IN ADDITION, VARIATION OF THE FILM COEFFICIENT WITH FLOW IS ACCOUNTED FOR AS WELL AS THE PROVISION FOR FLOW DECAY AND AFTERGLOW HEATING. SCRAMS CAN BE INITIATED BY DELAYED SIGNALS FROM INSTRUMENTS WHICH SENSE ANY QUANTITY CALCULATED, E.G., POWER, INVERSE PERIOD OR TEMPERATURE. GENERALIZED FEEDBACK EQUATIONS ARE USED TO PROVIDE FLEXIBILITY IN THE MODELS THAT REPRESENT MULTICHANNEL HEAT TRANSFER INCLUDING CONDUCTION AND CONVECTION, ENERGY, PRESSURE AND OTHER PHENOMENON. THE REACTIVITY EQUATION IS ALSO GENERALIZED. THE REACTIVITY FEEDBACK COEFFICIENTS CAN BE CONSTANT OR VARY AS THE SQUARE ROOT OR RECIPROCAL OF TEMPERATURE. FURTHERMORE ANY FEEDBACK VARIABLE CAN BE USED TO INITIATE A REACTIVITY SCRAM, EACH WITH A UNIQUE DELAY TIME.
4. METHOD OF SOLUTION - THE NUMERICAL TECHNIQUE USED TO INTEGRATE THE NEUTRON AND FEEDBACK DIFFERENTIAL EQUATIONS IS THAT DEVELOPED BY E. R. COHEN AS PREVIOUSLY USED IN THE AIREK CODES. AN IMPROVED INTERVAL SWITCHING TECHNIQUE ALLOWS RAPID CALCULATIONS WITH PRE-DETERMINED ACCURACY.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
15 DELAYED NEUTRON PRECURSOR GROUPS.
100 FEEDBACK VARIABLES WITH ANY VARIABLE CONNECTED TO NO MORE THAN 8 OTHERS.
39 FEEDBACK VARIABLES PRINTED.
6. TYPICAL RUNNING TIME - THE PROGRAM REQUIRES ONE TO TWO MINUTES FOR A PROBLEM WITH SIX DELAYED NEUTRON PRECURSOR GROUPS AND 50 FEEDBACK VARIABLES (INCLUDING CRT) ON THE IBM7094.
7. UNUSUAL FEATURES OF THE PROGRAM -
 - (A) SPECIAL PROVISION IS MADE FOR REACTOR START-UP PROBLEMS RESULTING IN LARGE REDUCTION IN RUNNING TIME.
 - (B) MUCH OF THE REQUIRED INPUT DATA IS PRE-SET BUT CAN BE CHANGED IF DESIRED.
 - (C) ADDRESSABLE INPUT DATA IS USED SO THAT ON MULTIPLE CASES, ONLY CHANGES NEED BE SPECIFIED.
 - (D) A RESTART FEATURE IS PROVIDED WHEREIN RESTART CARDS ARE PUNCHED UPON ABNORMAL PROBLEM TERMINATION AND/OR ON AN INPUT OPTION.
 - (E) EXTENSIVE PRINTED AND GRAPHICAL DISPLAYS ARE PROVIDED AS FOLLOWS - POWER, INVERSE PERIOD, REACTIVITY AND ANY 39 FEEDBACK VARIABLES. PRINTING AND DISPLAY OF FEEDBACK

7. UNUSUAL FEATURES OF THE PROGRAM (CONTINUED)
VARIABLES IS UNDER USER CONTROL AND THE LATTER CAN
BE GROUPED ON CRT FRAMES AS DESIRED.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED APRIL 1965.
IBM7094 VERSION SUBMITTED MARCH 1965.
IBM360 VERSION SUBMITTED FEBRUARY 1967, DELETED FEBRUARY
1968, REPLACED BY AIROS2A (ACC ABSTRACT 326).
10. REFERENCES - R. A. BLAINE AND R. F. BERLAND, AIROS - A DIGITAL
SIMULATOR FOR POWER REACTOR DYNAMICS, NAA-SR-9943, AUGUST 1964.
R. A. BLAINE, MODIFICATIONS TO AIROS, ATOMICS INTER-
NATIONAL INTERNAL LETTER, OCTOBER 1, 1964.
M. HOFFMAN AND W. A. RHOADES, AICRT 3, A GENERAL CODE
FOR DISPLAY OF DIGITAL DATA, NAA-SR-MEMO-9069, OCTOBER 1963.
11. MACHINE REQUIREMENTS - 32K IBM7094, SC-4020 GRAPHICAL DISPLAY
DEVICE
12. PROGRAMMING LANGUAGES USED - FORTRAN II + MAP
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
IBSYS.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
IF AN SC-4020 GRAPHICAL DISPLAY DEVICE IS NOT AVAILABLE, THE
AICRT3 DISPLAY ROUTINE CAN BE REWRITTEN, OTHERWISE LINK 2 CAN BE
DELETED AND AIROS CAN BE RUN AS A NON-OVERLAY JOB WITHOUT CRT.
THE NAA SC-4020 (IBSYS) SUBROUTINE PACKAGE IS AVAILABLE THROUGH -
UAIDE LIBRARIAN
C/O STROMBERG-CARLSON
P. O. BOX 2449
SAN DIEGO, CALIFORNIA
AIROS2A (ACC ABSTRACT 326) IS RECOMMENDED FOR 360 USERS.
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (7094-2000 CARDS)
SAMPLE PROBLEM (7094-173 CARDS)
REFERENCE REPORT AND LETTER

17. CATEGORY - E
KEYWORDS - SPACE-INDEPENDENT KINETICS, REACTIVITY, FEEDBACK, HEAT
TRANSFER

7. UNUSUAL FEATURES OF THE PROGRAM (CONTINUED)
COEFFICIENTS PUNCHED BY ABACUS 2 WILL USUALLY NOT COINCIDE WITH THE NUMBER REQUIRED IN THE INPUT OF NEARREX.
8. RELATED AND AUXILIARY PROGRAMS - ABACUS2 PRODUCES TRANSMISSION COEFFICIENTS THAT CAN BE USED BY NEARREX.
9. STATUS - ABSTRACT FIRST DISTRIBUTED FEBRUARY 1966.
CDC3600 VERSION SUBMITTED AUGUST 1965, SAMPLE PROBLEM EXECUTED BY ACC.
10. REFERENCE - P. A. MOLDAUER, C. A. ENGELBRECHT, AND G. J. DUFFY, NEARREX, A COMPUTER CODE FOR NUCLEAR REACTION CALCULATIONS, ANL-6978, DECEMBER 1964.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGES USED - FORTRAN 63 (CDC3600)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE (CDC3600).
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
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16. MATERIAL AVAILABLE -
SOURCE DECK (1041 CARDS)
BINARY DECK (400 CARDS)
SAMPLE PROBLEM (19 CARDS)
REFERENCE REPORT
17. CATEGORY - A
KEYWORDS - COMPOUND NUCLEUS, CROSS SECTIONS, ELASTIC SCATTERING, INELASTIC SCATTERING, CAPTURE, FISSION

1. NAME OR DESIGNATION OF PROGRAM - AISITE2/R153
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7094, CDC3600, IBM360
3. NATURE OF PHYSICAL PROBLEM SOLVED - AISITE2 IS LARGELY BASED ON METHODS PROPOSED BY THE AEC IN TID-14844 BUT DIFFERS IN CERTAIN OF THE ASSUMPTIONS AND MODELS. THE CODE AUTOMATICALLY VARIES ANY ONE OF 46 PARAMETERS SUCH AS REACTOR POWER, BUILDING LEAK RATE, IODINE CLEAN-UP RATE, AND HALOGEN FILTER EFFICIENCY, COMPUTING THE EXCLUSION AREA, AND LOW POPULATION BOUNDARY ZONES AS FUNCTIONS OF THAT PARAMETER. THE EDIT INCLUDES DOSE VS. DISTANCE DATA, FRACTIONAL CONTRIBUTION BY ISOTOPE GROUP TO THE INHALATION DOSE, AND CRITICAL DISTANCES PROVIDING BOTH PRINTED AND GRAPHICAL DATA. THREE MODELS ARE AVAILABLE FOR FISSION PRODUCT RELEASE WITH UP TO 4 LEVELS OF CONTAINMENT.
4. METHOD OF SOLUTION - ANALYTIC SOLUTIONS TO THE DIFFERENTIAL EQUATIONS ARE EVALUATED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 100 ISOTOPES
 - 20 DISTANCES
 - 10 VALUES OF VARIABLE PARAMETER
 - 8 ORGANS
 - 4 LEVELS OF CONTAINMENT
6. TYPICAL RUNNING TIME - ONE CASE WITH 10 DISTANCES AND 10 VALUES OF THE VARIABLE PARAMETER REQUIRES ABOUT 20 SECONDS FOR BOTH PRINTED AND GRAPHICAL OUTPUT.
7. UNUSUAL FEATURES OF THE PROGRAM - AISITE2 HAS BEEN OPTIMIZED TO REDUCE RUNNING TIME BY KEEPING TRACK OF THOSE PORTIONS OF THE CALCULATION WHICH NEED NOT BE RE-EVALUATED, I.E., ARE NOT AFFECTED BY THE VARIABLE PARAMETER. THIS OPTIMIZATION RESULTED IN UP TO A FACTOR OF 10 SAVINGS OVER THE EARLIER VERSION OF THE CODE.
8. RELATED AND AUXILIARY PROGRAMS - THIS PROGRAM IS A LATER VERSION OF THE PROGRAM, AISITE (NAA-SR-MEMO-9235).
9. STATUS - ABSTRACT FIRST DISTRIBUTED FEBRUARY 1966.
 - IBM7094 VERSION SUBMITTED APRIL 1965.
 - CDC3600 VERSION SUBMITTED OCTOBER 1965, SAMPLE PROBLEM EXECUTED BY ACC.
 - IBM360 VERSION SUBMITTED MARCH 1968.
10. REFERENCES - R. A. BLAINE AND E. L. BRAMBLETT, AISITE II, A DIGITAL COMPUTER PROGRAM FOR INVESTIGATION OF REACTOR SITING, NAA-SR-9982, SEPTEMBER 1964.
 - M. HOFFMAN AND W. A. RHCADDES, AICRT3, A GENERAL CODE FOR DISPLAY OF DIGITAL DATA, NAA-SR-MEMO 9069, OCTOBER 1963.

10. REFERENCES (CONTINUED)

R. A. BLAINE, ATOMICS INTERNATIONAL ENVIRONMENT
REPORT, MAY 1967.

D. H. LOVEGROVE, ANL-AMD PROGRAM LIBRARY REPORT R153,
OCTOBER 1965, AND MODIFICATIONS.

11. MACHINE REQUIREMENTS - 32K IBM7094 WITH SC-4020 GRAPHICAL DISPLAY
DEVICE, 256K BYTE IBM360

12. PROGRAMMING LANGUAGES USED - FORTRAN II AND FAP (IBM), FORTRAN 63
(CDC), FORTRAN IV (IBM360)

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
STANDARD IBM FORTRAN MONITOR (7090), SCOPE (3600), OS/360.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
IF AN SC-4020 GRAPHICAL DISPLAY DEVICE IS NOT AVAILABLE, DUMMY
ROUTINES CAN BE USED FOR SUBROUTINES GRAPH AND AICRT3, OR THESE
ROUTINES CAN BE REWRITTEN. THE NAA SC-4020 SUBROUTINE PACKAGE
IS AVAILABLE THROUGH -

UAIDE LIBRARIAN
C/O STROMBERG-CARLSON
P. O. BOX 2449
SAN DIEGO, CALIFORNIA

15. NAME AND ESTABLISHMENT OF AUTHORS -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (7094-2413 CARDS, 3600-1781 CARDS, 360-2833 CARDS)
BINARY DECK (3600-181 CARDS)
SAMPLE PROBLEMS (7094-99 CARDS, 3600-100 CARDS, 360-99 CARDS)
REFERENCE REPORTS

17. CATEGORY - G

KEYWORDS - DOSE RATES, DISTANCE, CONTAINMENT, FISSION PRODUCTS

1. NAME OR DESIGNATION OF PROGRAM - 2DF
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC1604A
3. NATURE OF PHYSICAL PROBLEM SOLVED - 2DF IS A TWO-DIMENSIONAL MULTIGROUP PROGRAM WRITTEN IN FORTRAN FOR SOLVING THE NEUTRON TRANSPORT EQUATION USING THE SN METHOD. THE PROGRAM CAN DETERMINE THE REAL OR ADJOINT SOLUTION FOR X-Y, R-Z, OR R-THETA GEOMETRY. ISOTROPIC OR A FORM OF LINEAR ANISOTROPIC SCATTERING MAY BE CONSIDERED. VARIOUS BOUNDARY CONDITIONS ARE ALLOWED. THE PROGRAM ALSO CONTAINS A NUMBER OF SEARCH OPTIONS WHEREBY ONE CAN VARY DIMENSIONS OR CONCENTRATIONS TO ARRIVE AT A PREDETERMINED EIGENVALUE. A DISTRIBUTED SOURCE MAY BE SPECIFIED. A LIBRARY OF CROSS SECTIONS IS AVAILABLE ON MAGNETIC TAPE. CROSS SECTIONS MAY BE READ FROM THE LIBRARY TAPE AND/OR FROM CARDS.
4. METHOD OF SOLUTION -
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - STORAGE RESTRICTIONS ARE SPECIFIED BY A COMPLICATED EQUATION INVOLVING S(N) ORDER, NUMBER OF MATERIALS, MESH INTERVALS, AND ENERGY GROUPS, AND SPECIFIED BOUNDARY CONDITIONS.
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - 2DF IS A TWO-DIMENSIONAL VERSION OF THE DTF PROGRAM (ACC ABSTRACT 144).
9. STATUS - ABSTRACT FIRST DISTRIBUTED FEBRUARY 1966.
CDC1604 VERSION SUBMITTED JANUARY 1966.
10. REFERENCES - UNITED NUCLEAR CORPORATION NOTES ON 2DF INPUT DATA DECK ARRANGEMENT, 2DF PROBLEM DECK ARRANGEMENT, OPERATING INSTRUCTIONS, 2DF INPUT FORMS.
2DF CROSS SECTION INPUT, PHYS/MATH MEMO 3816,
OCTOBER 23, 1964.
MAXIMUM SIZE 2DF PROBLEM, PHYS/MATH MEMO 3703,
SEPTEMBER 9, 1964.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN 63
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - COOP MONITOR.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

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15. NAME AND ESTABLISHMENT OF AUTHOR -

MARTIN SHAPIRO
UNITED NUCLEAR CORPORATION
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16. MATERIAL AVAILABLE -

SOURCE DECK (2590 CARDS)

REFERENCE NOTES AND INTER-OFFICE MEMORANDA

17. CATEGORY - C

KEYWORDS - SN METHOD, 2-DIMENSIONAL, MULTIGROUP, X-Y, R-Z, R-
THETA, CRITICALITY SEARCHES, ANISOTROPIC SCATTERING

1. NAME OR DESIGNATION OF PROGRAM - FORE/FORE2
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - PHILCO-2000, GE635, IBM360
3. NATURE OF PHYSICAL PROBLEM SOLVED - FORE CALCULATES REACTOR POWER AND TEMPERATURES OF FUEL, COOLANT, CLAD, AND STRUCTURE AS FUNCTIONS OF TIME IN RESPONSE TO A PROGRAMMED REACTIVITY INSERTION SPECIFIED AS A SERIES OF RAMPS. TEMPERATURE PROFILES ARE COMPUTED AT SPECIFIED AXIAL POSITIONS FOR AN AVERAGE CHANNEL AND FOR THE PEAK POWER (CENTRAL) CHANNEL. THE HEAT OF FUSION ACCOMPANYING FUEL MELTING IS TAKEN INTO ACCOUNT. FEEDBACK REACTIVITY MECHANISMS THAT RESPOND TO CHANGES OF TEMPERATURES INCLUDE THE FUEL DOPPLER EFFECT AND THERMAL EXPANSION OF THE CORE (AND AXIAL BLANKET OR REFLECTOR) MATERIALS. FORE2 IS A COUPLED THERMAL HYDRAULICS-POINT KINETICS DIGITAL COMPUTER CODE DESIGNED TO CALCULATE SIGNIFICANT REACTOR PARAMETERS UNDER STEADY-STATE CONDITIONS, OR AS FUNCTIONS OF TIME DURING TRANSIENTS. THE TRANSIENTS MAY RESULT FROM A PROGRAMMED REACTIVITY INSERTION OR A POWER CHANGE. VARIABLE INLET COOLANT FLOW RATE AND TEMPERATURE ARE CONSIDERED. THE CODE CALCULATES THE REACTOR POWER, THE INDIVIDUAL REACTIVITY FEEDBACKS, AND THE TEMPERATURE OF COOLANT, CLADDING, FUEL, STRUCTURE, AND ADDITIONAL MATERIAL FOR UP TO SEVEN AXIAL POSITIONS IN THREE CHANNEL TYPES WHICH REPRESENT RADIAL ZONES OF THE REACTOR. THE HEAT OF FUSION ACCOMPANYING FUEL MELTING, THE LIQUID METAL VOIDING REACTIVITY, AND THE SPATIAL AND THE TIME VARIATION OF THE FUEL CLADDING GAP COEFFICIENT DUE TO CHANGES IN GAP SIZE ARE CONSIDERED.
4. METHOD OF SOLUTION - FORE2 INPUT CONSISTS OF PROPERTY DATA, GEOMETRY, POWER AND FLOW DISTRIBUTION FACTORS, EXTERNAL TIME VARYING FUNCTIONS, EXPERIMENTAL COEFFICIENTS, AND TERMINATION DATA. THE DIFFERENTIAL EQUATIONS FOR FLUID FLOW, HEAT TRANSFER, AND POINT NEUTRONICS ARE SOLVED BY EXPLICIT FINITE-DIFFERENCE PROCEDURES.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - REACTOR EXCURSIONS WHICH CAN BE CALCULATED ARE RESTRICTED TO THOSE TRANSIENTS IN WHICH THE REACTOR IS NOT SUBSTANTIALLY DESTROYED. AS A GENERAL RULE, CHANGES IN REACTOR GEOMETRY AND COMPOSITION DURING AN EXCURSION ARE LIMITED TO THOSE CASES IN WHICH THE REACTIVITY EFFECTS OF THE CHANGES MAY BE CONSIDERED AS SMALL PERTURBATIONS OF THE INITIAL SYSTEM. THUS, ACCIDENTS INVOLVING LARGE-SCALE DISASSEMBLY AND BULK MELTDOWN OF A CORE ARE NOT COVERED BY FORE OR FORE2. UP TO SIX DELAYED NEUTRON GROUPS MAY BE USED IN FORE. FORE2 IS VALID ONLY WHILE THE CORE RETAINS ITS INITIAL GEOMETRY.
6. TYPICAL RUNNING TIME - RUNNING TIME OF FORE IS PROPORTIONAL TO THE NUMBER OF TIME-STEPS, GENERALLY 2 TO 26 MINUTES. ON A 10 MICROSECOND COMPUTER FORE2 RUNNING TIME IS ON THE ORDER OF 3.5 SECONDS PER STEP, FOR A 1 MICROSECOND COMPUTER 0.5 SECOND PER STEP.

7. UNUSUAL FEATURES OF THE PROGRAM - UNUSUAL FEATURES OF FORE2 INCLUDE -
 - (A) VARIABLE CLAD-FUEL GAP COEFFICIENT MODEL.
 - (B) SIMULATION OF LIQUID METAL VOIDING.
 - (C) UNLIMITED WRAPUP AND RESTART CAPABILITY WITH CHANGES IN INPUT PERMITTED BEFORE RESTART.
 - (D) DETERMINATION OF MAXIMUM STABLE TIME-STEP BY CODE.
 - (E) SEPARATION OF POWER INTO PROMPT AND TOTAL (INCLUDES DECAY HEAT).
 - (F) CAPABILITY TO USE ANY SINGLE PHASE COOLANT.
 - (G) INCLUSION OF FUEL BOWING AND CORE DEFORMATION DUE TO AXIAL PRESSURE DIFFERENCE.
 - (H) ABILITY TO HANDLE CORRECTLY THE VARIATION OF COOLANT FLOW RATE IN CHANNELS WITH ORIFICING DURING A COASTDOWN ACCIDENT.
8. RELATED AND AUXILIARY PROGRAMS - THE FORE PROGRAM CONSISTS OF THE MAIN PROGRAM FORE WHICH HANDLES BOOKKEEPING, ERROR CHECKING, AND RESTARTING AND SEVEN SUBROUTINES - INPUT, INIT, POWT, COPH, FEED, TERM, AND OUTPUT. FORE2 CONSISTS OF THREE INDEPENDENT CODES -
 - (A) EEEE PERFORMS ALL CALCULATIONS AND GIVES A CONDENSED OR SHORT EDIT OF THE RESULTS.
 - (B) FEDIT GIVES A DETAILED OR LONG EDIT OF THE RESULTS.
 - (C) FRCURV PERFORMS CURVE PLOTTING OF THE OUTPUT.
9. STATUS - ABSTRACT FIRST DISTRIBUTED FEBRUARY 1966.
 FORE, PHILCO-2000 VERSION, SUBMITTED NOVEMBER 1965.
 FORE2, GE635 VERSION, SUBMITTED JULY 1967.
 FORE2, IBM360 VERSION, SUBMITTED FEBRUARY 1968.
10. REFERENCES - P. GREEBLER, D. B. SHERER, N. H. WALTON, FORE, A COMPUTATIONAL PROGRAM FOR THE ANALYSIS OF FAST REACTOR EXCURSIONS, GEAP-4090, OCTOBER 16, 1962, PLUS ERRATA AND ADDENDA, EA2, FEBRUARY 10, 1965, EA3, JUNE 22, 1965, AND EA4, DECEMBER 22, 1965.
 T. J. HEAMES, ADAPTATION OF THE GENERAL ELECTRIC FAST OXIDE REACTOR EXCURSION CODE NUMBER TWO, FORE-II, TO OPERATING SYSTEM 360 MODEL 50/75, ANL REACTOR PHYSICS DIVISION MEMO, FEBRUARY 1968.
 J. N. FOX, B. E. LAWLER, AND H. R. BUTZ, FORE II, A COMPUTATIONAL PROGRAM FOR THE ANALYSIS OF STEADY-STATE AND TRANSIENT REACTOR PERFORMANCE, GEAP-5273, SEPTEMBER 1966.
 FORE2 SUBROUTINE DESCRIPTIONS INCLUDING -
 NELS J. ANDERSON, JR., IDFTAP, A FORTRAN IV FUNCTION SUBPROGRAM, S6T001, JANUARY 15, 1965.
 BITA, L5T001, DECEMBER 3, 1964.
 NELS J. ANDERSON, JR., FORTRAN IV SUBROUTINE ERRSYS, S4T001, JANUARY 15, 1965.
 FLEXO, J3T001, JANUARY 21, 1965.
 FLTBCC, L2T006.
 LISTFORM, S6T002, FEBRUARY 12, 1964.
 NFXBCC, L2T004, NOVEMBER 16, 1965.

10. REFERENCES (CONTINUED)

C. T. EHLERS, NSCRT, P5T001, OCTOBER 29, 1964.
JOHN W. WEIL, PLOTRC, K5T003.
WALTER A. DUNCAN, VAMPIR, K5F009, APRIL 25, 1966.
A. E. SCADDEN, ZERO, DIVISION OVERFLOW MODIFIER,

Q4T001.

11. MACHINE REQUIREMENTS - 32K MEMORY (FORE), 3 FILES AND 40K MEMORY (FORE2-635)

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SMART, THE GE-APED MONITOR, GECOS (635), OS/360.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - FORE REQUIRES AS INPUT A MINIMUM AND MAXIMUM STEP-SIZE WHICH WILL PLACE A LIMIT ON THE DOUBLING OR HALVING PROCESS, AND AN INITIAL STEP-SIZE WITH WHICH TO BEGIN THE CALCULATION. FORE2 CALLS SEVERAL SYSTEMS SUBROUTINES WHICH ARE NOT INCLUDED WITH THE PROGRAM. HOWEVER, WRITEUPS OF THESE SUBROUTINES ARE INCLUDED WITH THE CODE PACKAGE.

15. NAME AND ESTABLISHMENT OF AUTHORS -

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ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC COMPANY
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FORE2

635 J. N. FOX, B. E. LAWLER, AND H. R. BUTZ
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SUNNYVALE, CALIFORNIA

360 T. J. HEAMES
REACTOR PHYSICS DIVISION
ARGONNE NATIONAL LABORATORY
9700 SOUTH CASS AVENUE
ARGONNE, ILLINOIS 60439

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECKS (FORE 2000-2687 CARDS, FORE2 635-8763 CARDS,
360-8461 CARDS, FEDIT 635-1148 CARDS, 360-1465
CARDS, 360-917 CARDS, FRCURV 635-2305 CARDS)

SAMPLE PROBLEMS (FORE 2000-61 CARDS, FORE2 635-139 CARDS,
360-128 CARDS)

REFERENCE REPORTS, ERRATA AND ADDENDA, MEMO, FORE2 SUBROUTINE
DESCRIPTIONS

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17. CATEGORY - F

KEYWORDS - EXCURSIONS, POWER DISTRIBUTION, TEMPERATURE, FEEDBACK,
KINETICS, FUELS, COOLANTS, FLUID FLOW, HEAT TRANSFER,
EEEE CODES, FEDIT CODES, FRCURV CODES

1. NAME OR DESIGNATION OF PROGRAM - GAMBLE4/GAMBLE5
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7044/UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE HOMOGENEOUS TWO-DIMENSIONAL MULTIGROUP DIFFUSION THEORY EQUATIONS WITH ARBITRARY GROUP-TO-GROUP SCATTERING AND ARBITRARY FISSION TRANSFER ARE SOLVED FOR HETEROGENEOUS ASSEMBLIES IN X-Y AND R-Z GEOMETRY. HOMOGENEOUS LOGARITHMIC BOUNDARY CONDITIONS ARE USED AT THE OUTER SURFACE OF THE ASSEMBLY AND AT THE SURFACE OF NON-DIFFUSION REGIONS. THE RESULTS INCLUDE THE GROUP AND POINT DEPENDENT NEUTRON FLUXES, THE POWER DISTRIBUTION, THE NEUTRON MULTIPLICATION FACTOR (K-EFFECTIVE), AND A DETAILED NEUTRON BALANCE.
4. METHOD OF SOLUTION - THE MULTIGROUP DIFFUSION THEORY EQUATIONS ARE APPROXIMATED BY FIVE-POINT DIFFERENCE EQUATIONS FOR AN ARBITRARY NON-UNIFORM MESH-GRID. THE SYSTEM OF DIFFERENCE EQUATIONS IS SOLVED BY AN EXTENSION OF THE POWER METHOD TO FIND THE EIGENVECTOR (NEUTRON FLUX) AND THE EIGENVALUE (K-EFFECTIVE). SUCCESSIVE LINE OVERRELAXATION IS APPLIED IN A SPECIAL FORM (EXPONENTIAL OVERRELAXATION) WHICH GUARANTEES THE NON-NEGATIVITY OF THE NEUTRON FLUX. COARSE MESH REBALANCING IS USED TO IMPROVE THE PRE-ASYMPTOTIC CONVERGENCE BEHAVIOR. A VARIATION OF AITKENS METHOD IS USED TO IMPROVE THE ASYMPTOTIC CONVERGENCE BEHAVIOR, IN ASSUMING ONLY ONE ERROR MODE.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
IBM7044 - MAXIMA OF -
 10 ENERGY GROUPS
 300 DIFFERENT MATERIAL REGIONS
 20,000 SPACE MESH POINTS
UNIVAC1108 - MAXIMA OF -
 10 ENERGY GROUPS
 255 DIFFERENT MATERIAL REGIONS
 20,000 SPACE MESH POINTS
6. TYPICAL RUNNING TIME - ON THE UNIVAC1108 A SEVEN-GROUP PROBLEM (THREE FAST GROUPS AND 4 THERMAL GROUPS) IN R-Z GEOMETRY WITH 2842 SPACE MESH POINTS TOOK 82 ITERATIONS ASSUMING A TIGHT CONVERGENCE CRITERION (MAXIMUM RELATIVE FLUX CHANGE LESS THAN 0.00007). THE TOTAL RUNNING TIME (INCLUDING EXTENSIVE OUTPUT) WAS 12 MINUTES.
7. UNUSUAL FEATURES OF THE PROGRAM -
 (A) THE COARSE MESH REBALANCING SCHEME MAKES THE SUCCESSFUL SOLUTION OF DIFFICULT PROBLEMS POSSIBLE FOR WHICH CERTAIN GROUP-MESH POINTS ARE BOTH STRONGLY AND WEAKLY COUPLED TO SOME OF THEIR NEIGHBORS (FOR EXAMPLE, HIGHLY NON-UNIFORM MESH SPACINGS OR MATERIAL PROPERTIES, AIRGAPS, CELL PROBLEMS WITH WEAK GROUP COUPLING, ETC.).
 (B) GOOD CONVERGENCE BEHAVIOR FOR LARGE POWER REACTOR PROBLEMS IS ACCOMPLISHED BY A COMBINATION OF COARSE MESH REBALANC-

7. UNUSUAL FEATURES OF THE PROGRAM (CONTINUED)
ING AND AN ASYMPTOTIC FLUX EXTRAPOLATION PROCEDURE WHICH ASSUMES A SINGLE ERROR MODE.
- (C) SIMULTANEOUS PERFORMANCE OF COMPUTATION AND DATA TRANSFER WITH VIRTUALLY NO DELAY CAUSED BY THE USE OF DRUM STORAGE.
 - (D) ABILITY TO DO EFFICIENT RESTARTS FOR LONGER-RUNNING PROBLEMS AND THE ABILITY TO ACCEPT A FLUX GUESS ON TAPE FROM A SIMILAR PROBLEM.
8. RELATED AND AUXILIARY PROGRAMS - THE PROGRAMMING EFFORT OF GAMBLE5 IS A MAJOR REVISION OF MOST OF GAMBLE4. MOST OF THE ESSENTIALS OF THE ITERATIVE TECHNIQUE USED HAVE BEEN ADOPTED FROM EXTERMINATOR (ACC ABSTRACT 156). IT SHOULD ALSO BE NOTED THAT -
- (A) THE PROGRAM USES A SPECIAL ASSEMBLY LANGUAGE PROGRAM CALLED CTTS FOR DRUM AND TAPE USE.
 - (B) THE PROGRAM CTTS USES A SPECIAL VERSION OF NTRAN WHICH PERMITS OVERWRITES OF SELECTED RECORDS IN A DRUM DATA FILE, AND
 - (C) THE PROGRAM DEPENDS STRONGLY ON SPECIAL FEATURES OF THE GULF GENERAL ATOMIC 1108 EXEC II OPERATING SYSTEM.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JULY 1966.
IBM7044 VERSION (GAMBLE4) SUBMITTED FEBRUARY 1965.
UNIVAC1108 VERSION (GAMBLE5) SUBMITTED JANUARY 1968.
10. REFERENCES - J. P. DORSEY, GAMBLE-4, A PROGRAM FOR THE SOLUTION OF THE MULTIGROUP NEUTRON DIFFUSION EQUATIONS IN TWO DIMENSIONS, WITH ARBITRARY GROUP SCATTERING, FOR THE IBM7044 FORTRAN IV SYSTEM, GA-6540, JULY 1, 1965.
J. P. DORSEY AND R. FROELICH, GAMBLE-5, A PROGRAM FOR THE SOLUTION OF THE MULTIGROUP NEUTRON-DIFFUSION EQUATIONS IN TWO DIMENSIONS, WITH ARBITRARY GROUP SCATTERING, FOR THE UNIVAC-1108 COMPUTER, GA-8188, DECEMBER 4, 1967.
R. FROELICH, A THEORETICAL FOUNDATION FOR COARSE MESH VARIATIONAL TECHNIQUES, PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON RESEARCH REACTOR UTILIZATION AND REACTOR MATHEMATICS, VOLUME 1, PAGES 219-237, MAY 2-4, 1967.
T. B. FOWLER, M. TOBIAS, AND D. VONDY, EXTERMINATOR, A MULTIGROUP CODE FOR SOLVING NEUTRON DIFFUSION EQUATIONS IN ONE AND TWO DIMENSIONS, ORNL-TM-842, JULY 1966.
11. MACHINE REQUIREMENTS - 32K IBM7044 WITH 11 TAPE UNITS ON TWO DATA CHANNELS AND A PERIPHERAL PRINTER. 65,536 WORDS OF CORE STORAGE (UNIVAC1108) AND 3 TAPE UNITS ON ONE DATA CHANNEL, 1,572,864 WORDS OF FH-88C DRUM STORAGE FROM ONE DATA CHANNEL, AND A PERIPHERAL PRINTER.
12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND MAP (IBM7044), FORTRAN IV AND ASSEMBLY LANGUAGE (UNIVAC1108)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - IBM7040/7044 OPERATING SYSTEM (16/32K), VERSION IX AND EXEC-II, GAX23 (UNIVAC1108).

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14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
ON THE IBM7044 THE USE OF A PREVIOUS SOLUTION AS INPUT AND AN
EFFICIENT RESTART PROCEDURE REQUIRES AN ADDITIONAL TAPE UNIT.
15. NAME AND ESTABLISHMENT OF AUTHORS -
J. P. DORSEY AND R. FROELICH
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (GAMBLE4 7044-15,933 CARDS, GAMBLE5 1108-16,043
CARDS)
BINARY DECK (GAMBLE4 7044-3652 CARDS)
SAMPLE PROBLEMS (GAMBLE4 7044-47 CARDS, GAMBLE5 1108-48 CARDS)
REFERENCE REPORTS, GA-654C AND GA-8188.
17. CATEGORY - C
KEYWORDS - DIFFUSION EQUATIONS, 2-DIMENSIONAL, MULTIGROUP, X-Y,
R-Z, REACTIVITY, FLUX DISTRIBUTION

REACTOR CODE ABSTRACT 265

8/68

1. NAME OR DESIGNATION OF PROGRAM - RSAC
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7040,44
3. NATURE OF PHYSICAL PROBLEM SOLVED - RSAC GENERATES A FISSION PRODUCT INVENTORY FROM A GIVEN SET OF REACTOR OPERATING CONDITIONS AND THEN COMPUTES THE EXTERNAL GAMMA DOSE, THE DEPOSITION GAMMA DOSE, AND/OR THE INHALATION-INGESTION DOSE TO CRITICAL BODY ORGANS AS A RESULT OF EXPOSURE TO THESE FISSION PRODUCTS. PROGRAM OUTPUT INCLUDES REACTOR OPERATING HISTORY, FISSION PRODUCT INVENTORY, DOSAGES, AND INGESTION PARAMETERS.
4. METHOD OF SOLUTION - THE FISSION PRODUCT INVENTORY GENERATED BY THE REACTOR OPERATING CONDITIONS AND THE INVENTORY REMAINING AT VARIOUS TIMES AFTER RELEASE ARE COMPUTED USING THE EQUATIONS OF W. RUBINSON IN JOURNAL OF CHEMICAL PHYSICS, VOL. 17, PAGES 542-547, JUNE 1949. THE EXTERNAL GAMMA DOSE AND THE DEPOSITION GAMMA DOSE ARE CALCULATED BY DETERMINING DISINTEGRATION RATES AS A FUNCTION OF SPACE AND TIME AND THEN INTEGRATING USING HERMITES NUMERICAL TECHNIQUES FOR THE SPATIAL DEPENDENCE. THE INHALATION-INGESTION DOSE IS DETERMINED BY THE TYPE AND QUANTITY OF ACTIVITY INHALED AND THE BIOLOGICAL RATE OF DECAY FOLLOWING INHALATION. THESE QUANTITIES ARE INTEGRATED WITH RESPECT TO TIME TO OBTAIN THE DOSAGE. THE INGESTION DOSE IS RELATED TO THE INHALATION DOSE BY AN INPUT CONSTANT.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 90 ISOTOPE CHAINS
 - 5 ISOTOPE PER CHAIN
 - 10 ENERGY GROUPS
 - 10 SETS OF LEAKAGE EXPONENTIALS
 - 25 DOWNWIND AND CROSSWIND POSITIONS EACH
6. TYPICAL RUNNING TIME - 18 MINUTES ARE REQUIRED FOR THE CALCULATION OF GAMMA, DEPOSITION, AND INHALATION-INGESTION DOSAGES AND CALCULATION OF FISSION PRODUCT INVENTORY FOR 2 DOWNWIND POSITIONS UTILIZING 90 ISOTOPE CHAINS.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED APRIL 1967.
 - IBM7040 (MAP) VERSION SUBMITTED JANUARY 1967, DELETED AUGUST 1968.
 - IBM7040 (FORTRAN) VERSION SUBMITTED AUGUST 1968.
10. REFERENCES - R. L. COATES AND N. R. HORTON, RSAC - A RADIOLOGICAL SAFETY ANALYSIS COMPUTER PROGRAM, IDC-17151, MAY 1966.

10. REFERENCES (CONTINUED)

L. C. RICHARDSON, USERS MANUAL FOR FORTRAN VERSION OF RSAC, A RADIOLOGICAL SAFETY ANALYSIS COMPUTER PROGRAM, IDO-17261, MAY 1968.

11. MACHINE REQUIREMENTS - 32K MEMORY

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND MAP

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - STANDARD IBSYS FOR IBM7040,44.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -

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ATOMIC ENERGY DIVISION
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IDAHO FALLS, IDAHO

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (2666 CARDS)
SAMPLE PROBLEM (971 CARDS)
REFERENCE REPORTS

17. CATEGORY - G

KEYWORDS - RADIOACTIVITY, CONTINUOUS RELEASE, INSTANTANEOUS
RELEASE, FISSION PRODUCTS, ATMOSPHERE, DOSE RATES

15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECK (928 CARDS)

OBJECT DECK (980 CARDS)

TEMPEST AND FORM DECIMAL LIBRARIES (7077 CARDS)

REFERENCE REPORTS (FLOWCHART VOLUME MAY BE REQUESTED ON A LOAN
BASIS ONLY)

17. CATEGORY - M

KEYWORDS - CROSS SECTIONS, LIBRARIES, PREPARATION, MAINTENANCE,
RETRIEVAL, FORM CODES, THREDES CODES, DATA MANAGEMENT

1. NAME OR DESIGNATION OF PROGRAM - FIGRO
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - FIGRO CALCULATES THE ONE-DIMENSIONAL STEADY-STATE TEMPERATURE DISTRIBUTION AND TOTAL FUEL SWELLING FOR METAL-CLAD, AXISYMMETRIC, BULK-OXIDE CYLINDRICAL FUEL ELEMENTS. THE FUEL PELLET MAY BE SOLID, ANNULAR, OR CONTAIN TWO RADIAL ZONES. OXIDE FUEL THERMAL CONDUCTIVITY IS A FUNCTION OF TEMPERATURE, DEPLETION, AND POROSITY. FUEL SWELLING IS A FUNCTION OF TEMPERATURE, DEPLETION, INTERNAL HYDROSTATIC PRESSURE, AND FISSIIONING RATE. EITHER THE CLAD SURFACE FLUX OR THE TEMPERATURE AT THE INSIDE RADIUS OF THE FUEL MAY BE SPECIFIED AS A BOUNDARY CONDITION FOR THE HEAT CONDUCTION EQUATION. THERMAL EXPANSION OF THE FUEL AND CLADDING IS ACCOUNTED FOR.
4. METHOD OF SOLUTION - FUEL SWELLING IS COMPUTED FROM EITHER THE GREENWOOD-SPEIGHT MODEL OR THE ASSUMPTION OF FIXED BUBBLE NUMBER AND SPACING. THERMAL CONDUCTIVITY OF THE CLADDING IS A FUNCTION OF TEMPERATURE. THAT OF THE FUEL IS CALCULATED FROM A FUNCTIONAL RELATIONSHIP INVOLVING TEMPERATURE, DEPLETION, POROSITY, AND MATERIAL PROPERTIES. HEAT TRANSFER AT GAPS AND INTERFACES IS DESCRIBED BY USER SPECIFIED HEAT TRANSFER COEFFICIENTS OR BY A MODEL BASED ON CONDUCTION THROUGH A PHYSICAL GAP WHOSE SIZE IS DETERMINED BY FUEL AND CLAD DIFFERENTIAL THERMAL EXPANSION AND BY FUEL SWELLING. THE GAP CONDUCTIVITY IS A FUNCTION OF TEMPERATURE AND GAS COMPOSITION.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - EACH FUEL ZONE MAY BE SUBDIVIDED INTO A MAXIMUM OF 100 CONCENTRIC EQUAL-THICKNESS ANNULI FOR THE CALCULATION OF TEMPERATURE AND FUEL GROWTH.
6. TYPICAL RUNNING TIME - THE ESTIMATED RUNNING TIME IS ONE SECOND FOR A SPECIFIED CLAD-SURFACE FLUX PROBLEM AND 10 SECONDS FOR A SPECIFIED INTERNAL FUEL TEMPERATURE PROBLEM.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - IN FUEL ELEMENT DESIGN STUDIES FIGRO IS USED TO SET BASIC TEMPERATURE PARAMETERS AND AS A SURVEY TOOL FOR COMPARING DEPLETION CAPABILITIES WITH CYGRO (ACC ABSTRACT 266) USED TO CONFIRM THE ANALYSIS OF FUEL ELEMENT BEHAVIOR.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
PHILCO 2000 VERSION SUBMITTED MARCH 1967, DELETED JUNE 1968.
CDC6600 VERSION SUBMITTED JUNE 1968.

10. REFERENCES - I. GOLDBERG, L. L. LYNN, AND C. D. SPHAR, FIGRO - FORTRAN IV DIGITAL COMPUTER PROGRAM FOR THE ANALYSIS OF FUEL SWELLING AND CALCULATION OF TEMPERATURE IN BULK-OXIDE CYLINDRICAL FUEL ELEMENTS, WAPD-TM-618, DECEMBER 1966.
L. A. WALDMAN, L. L. LYNN, AND I. GOLDBERG, FIGRO (ADDENDUM) - A CDC-6600 COMPUTER PROGRAM FOR THE ANALYSIS OF FUEL SWELLING AND CALCULATION OF TEMPERATURE IN BULK-OXIDE CYLINDRICAL FUEL ELEMENTS (LWBR-LSBR DEVELOPMENT PROGRAM), WAPD-TM-618 ADDENDUM I, OCTOBER 1967.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 2.0.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - FIGRO USES THE INPF PACKAGE DESCRIBED IN REFERENCE 3.
15. NAME AND ESTABLISHMENT OF AUTHORS -
L. L. LYNN, L. A. WALDMAN, I. GOLDBERG,
AND C. D. SPHAR
WESTINGHOUSE ELECTRIC CORPORATION
BETTIS ATOMIC POWER LABORATORY
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122
16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (2616 CARDS)
SAMPLE PROBLEM (23 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS AND ADDENDUM
17. CATEGORY - H
KEYWORDS - TEMPERATURE DISTRIBUTION, FUEL ELEMENTS, CYLINDERS,
THERMAL EXPANSION, DEPLETION, HEAT TRANSFER, FUEL
SWELLING, HEAT CONDUCTION

1. NAME OR DESIGNATION OF PROGRAM - THREDES
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360
3. NATURE OF PHYSICAL PROBLEM SOLVED - THREDES IS A SCIENTIFIC APPLICATIONS PROGRAMMING SYSTEM. INCORPORATED IN THIS SYSTEM ARE THE NECESSARY MODULES TO PERFORM PARAMETRIC DESIGN STUDIES OF THERMAL REACTORS INCLUDING THE THERMAL CELL HOMOGENIZATION (BAM - ACC ABSTRACT 108), THE FAST SPECTRUM CALCULATION (FORM - ACC ABSTRACT 51), REACTOR DIFFUSION THEORY (FOG - ACC ABSTRACT 28), AND ZERO-DIMENSIONAL BURNUP (KINDLE) CALCULATIONS. THESE MODULES CAN BE USED IN CONJUNCTION WITH ONE ANOTHER OR INDIVIDUALLY.
4. METHOD OF SOLUTION -
 - BAM - ASSUMES SEPARABILITY OF SPACE AND ENERGY - ITERATES BETWEEN TEMPEST-II TO OBTAIN WIGNER-WILKINS, WILKINS OR MAXWELLIAN AVERAGED CROSS SECTIONS AND THE CYLINDRICAL CELL CODE TO OBTAIN THERMAL DISADVANTAGE FACTORS. PHYSICS DATA IS MADE AVAILABLE FOR USE IN FORM AND FOG OR KINDLE.
 - FORM - SOLVES THE FOURIER TRANSFORM OF THE BOLTZMANN EQUATION FOR THE FAST SPECTRUM AND PREPARES PHYSICS DATA FOR USE IN FOG OR KINDLE.
 - FOG - SOLVES THE 4 GROUP DIFFUSION THEORY APPROXIMATION TO THE ONE-DIMENSIONAL TRANSPORT EQUATION USING A STANDARD FINITE-DIFFERENCE SCHEME.
 - KINDLE - PERFORMS AN ANALYTIC SOLUTION OF ISOTOPE EQUATIONS AND REACTIVITY EQUATIONS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - BAM -
 - 10 REGIONS
 - 100 SPACE POINTS
 - 20 ISOTOPES PER REGION
 - FORM -
 - 18 DIFFERENT ISOTOPES
 - FOG -
 - 4 GROUPS
 - 238 SPACE POINTS
 - KINDLE -
 - 2 GROUPS
6. TYPICAL RUNNING TIME - 5 TO 10 MINUTES ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM -
 - (A) EMPHASIS ON MULTIPLE CASES WITH ONLY CHANGES SPECIFIED TO ALLOW RAPID PARAMETER SURVEYS.
 - (B) ABILITY TO USE MODULES INDIVIDUALLY.
 - (C) USE OF MODULES AS AN INTEGRATED SYSTEM TO PREPARE DATA FOR ONE ANOTHER.

REACTOR CODE ABSTRACT 278

6/68

1. NAME OR DESIGNATION OF PROGRAM - WHAM/WATER-HAMMER
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC3800,3600,6600/IBM360
3. NATURE OF PHYSICAL PROBLEM SOLVED - WHAM IS USED TO CALCULATE PRESSURE, VELOCITY, AND FORCE TRANSIENTS IN THE LIQUID FILLED PIPING NETWORKS. IT CAN BE APPLIED TO MULTI-LOOP, COMPLEX PIPING NETWORKS CONSISTING OF ANY NUMBER OF DEAD ENDS, ELBOWS, ORIFICES, MULTIPLE BRANCH TEES, CHANGES OF FLOW PASSAGE CROSS SECTIONS, CHECK VALVES, PUMPS, PRESSURIZERS OR TANKS, AND EXIT VALVES. HYDRAULIC LOSSES ARE CONSIDERED. TRANSIENTS CAN BE INITIATED EITHER, BY CLOSURE OR BY OPENING OF ONE OR MORE EXIT VALVES (OR BY THE SYSTEM RUPTURE AT ONE OR MORE PLACES), OR BY A PRESCRIBED CHANGE IN THE TANK (OR PRESSURIZER) PRESSURE, OR BY THE PUMP POWER FAILURE. WHAM IS AN EXTENSION OF THE PROGRAM WATER-HAMMER. WATER-HAMMER ANALYZES HYDRAULIC TRANSIENTS IMPOSED BY THE SUDDEN DISCHARGE OF LIQUID FROM THE PRIMARY COOLANT LOOP, BEFORE ANY GENERATION OF VAPOR HAS TAKEN PLACE. EFFECTS OF HYDRAULIC LOSSES ARE NEGLECTED, THE LOOP PRESSURIZER IS ASSUMED TO REMAIN CONSTANT AT INITIAL PRESSURE AND THE PRESENCE OF A PUMP IS IGNORED. IT WAS DESIGNED FOR THE EARLY BLOWDOWN ANALYSIS FOR LOFT.
4. METHOD OF SOLUTION - THE UNSTEADY FLUID MOTION IS DESCRIBED BY THE SET OF MASS, MOMENTUM, AND ENERGY CONSERVATION LAWS, THE EQUATION OF STATE FOR THE FLUID, AND THE INITIAL AND BOUNDARY CONDITIONS FOR ALL DEPENDENT VARIABLES. THE WELL-KNOWN WATER-HAMMER EQUATIONS FOR THE LIQUID-FILLED PIPES ARE THEN EVALUATED BY USE OF THE WAVE SUPERPOSITION METHOD. THE FRICTIONLESS CASE MAY BE CONSIDERED IF DESIRED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
100 LEGS
50 NODES PER LEG
6. TYPICAL RUNNING TIME - TYPICAL RUNNING TIME IS 15 TO 20 MINUTES.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
WATER-HAMMER
CDC3600 VERSION SUBMITTED MAY 1967, REPLACED BY WHAM
DECEMBER 1967.
IBM360 VERSION SUBMITTED JULY 1967.
WHAM
CDC3800 VERSION SUBMITTED DECEMBER 1967, SAMPLE PROBLEM
EXECUTED ON CDC3600 BY ACC.

10. REFERENCES - STANISLAV FABIC, COMPUTER PROGRAM WHAM FOR CALCULATION OF PRESSURE, VELOCITY, AND FORCE TRANSIENTS IN LIQUID FILLED PIPING NETWORKS, 67-49-R, NOVEMBER 1967.
STANISLAV FABIC, EARLY BLOWDOWN (WATER-HAMMER) ANALYSIS FOR LOSS-OF-FLUID TEST FACILITY, KAISER ENGINEERS, 65-28-RA, JUNE 1965, REVISED APRIL 1967.
11. MACHINE REQUIREMENTS - 64K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE (3600) AND OS/360.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - TURBINE OPERATION RANGE IS NOT CONSIDERED BECAUSE IT IS ASSUMED THAT EACH PUMP IS PROVIDED WITH A CHECK VALVE. ALL PUMPS, ALL CHECK VALVES, ALL TANKS, AND ALL EXIT VALVES MUST OPERATE IN ESSENTIALLY THE SAME EXTERNALLY PRESCRIBED MANNER IF MORE THAN ONE OF EACH OF THESE ITEMS ARE PRESENT. WHAM IS LIMITED TO THE LIQUID FILLED SYSTEMS. HOWEVER, TIE-IN POINTS HAVE BEEN BUILT INTO THE PROGRAM TO ENABLE FUTURE TESTING OF THE FLUID STATE AND INCORPORATION OF ROUTINES WHICH WOULD CONSIDER THE PHASE CHANGE AND A TWO-PHASE FLOW.
15. NAME AND ESTABLISHMENT OF AUTHOR -
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KAISER ENGINEERS
KAISER INDUSTRIES CORPORATION
KAISER CENTER
300 LAKESIDE DRIVE
OAKLAND, CALIFORNIA 94604

360 M. DUSZYNSKI
UNIVERSITY OF GEORGIA COMPUTER CENTER
ATHENS, GEORGIA 30601
16. MATERIAL AVAILABLE -
SOURCE DECKS (3800-723 CARDS, 360-471 CARDS)
BINARY DECK (3800-333 CARDS)
SAMPLE PROBLEMS (3800-27 CARDS, 360-74 CARDS)
REFERENCE REPORTS
17. CATEGORY - G
KEYWORDS - ACCIDENTS, COOLANTS, BLOWDOWN, LIQUIDS, LCFT REACTORS, HEAT TRANSFER, WATER HAMMER CODES

REACTOR CODE ABSTRACT 280

12/67

1. NAME OR DESIGNATION OF PROGRAM - M0807
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - M0807 SOLVES THE TWO-DIMENSIONAL FIXED-SOURCE DIFFUSION EQUATION FOR THE ABSORPTION AND REMOVAL MACROSCOPIC CROSS SECTIONS REQUIRED TO YIELD A SPECIFIED REACTION RATE DISTRIBUTION.
4. METHOD OF SOLUTION - THE SPATIAL FLUX SOLUTION IS OBTAINED ITERATIVELY BY A METHOD TERMED EXTRAPOLATED SIMULTANEOUS ITERATION AND THE FITTED DIFFUSION CONSTANTS ARE OBTAINED BY THE STRAIGHT POWER METHOD. THE APPROXIMATION TO THE FLUX SOLUTION IS MADE USING THE FOUR-POINT DIFFERENCE EQUATIONS IN RECTANGULAR GEOMETRY AND THE SEVEN-POINT EQUATIONS ARE USED FOR THE 60 DEGREE PARALLELOGRAM GEOMETRY.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - TWO GEOMETRIES ARE PERMITTED, RECTANGULAR AND 60 DEGREE PARALLELOGRAM. IN RECTANGULAR GEOMETRY 70 X 70 MESH POINTS ARE PERMITTED AND 60 X 60 FOR PARALLELOGRAM GEOMETRY. AS MANY AS 99 COMPOSITIONS ARE ALLOWED PER PROBLEM.
6. TYPICAL RUNNING TIME - ABOUT TWO MINUTES ARE REQUIRED FOR A 30 X 30 MESH PROBLEM HAVING 40 COMPOSITIONS.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - THE FITTED DIFFUSION CONSTANTS ARE USUALLY USED AS INPUT TO THE PDQ7 COMPUTER PROGRAM (ACC ABSTRACT 275) FOR AN EIGENVALUE CALCULATION.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED MARCH 1968.
10. REFERENCES - C. H. RUTHERFORD, M0807 - A DIFFUSION THEORY FITTING PROGRAM USING FORTRAN-IV, WAPD-TM-671, SEPTEMBER 1967.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
W. R. CADWELL, PDQ-7 REFERENCE MANUAL, WAPD-TM-678, JANUARY 1967.
11. MACHINE REQUIREMENTS - 64K CORE STORAGE, CARD READER, PRINTER, AND ONE AUXILIARY STORAGE UNIT
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCCPE 2.0.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -

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P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION

MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (803 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS

17. CATEGORY - B

KEYWORDS - 2-DIMENSIONAL, ABSORPTION, CAPTURE, CROSS SECTIONS,
REACTION RATES, CELL CALCULATION

REACTOR CODE ABSTRACT 281

12/67

1. NAME OR DESIGNATION OF PROGRAM - RABBLE/WLIB/FLAT
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC3600
3. NATURE OF PHYSICAL PROBLEM SOLVED - RABBLE COMPUTES EFFECTIVE CROSS SECTIONS FOR ABOVE THERMAL ENERGIES BASED ON RESOLVED SINGLE-LEVEL RESONANCE PARAMETERS FOR INFINITE HOMOGENEOUS OR HETEROGENEOUS SYSTEMS.
4. METHOD OF SOLUTION - THE CELL IS DIVIDED INTO REGIONS, AND THE ENERGY SPAN IS DIVIDED INTO MANY NARROW GROUPS OF EQUAL LETHARGY WIDTH. REGIONAL REACTION RATES ARE COMPUTED FROM SLOWING DOWN SOURCES, FIRST FLIGHT ESCAPE AND TRANSMISSION PROBABILITIES AND INTERFACE CURRENTS THAT ARE OBTAINED FROM A STATEMENT OF NEUTRON CONSERVATION.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
5 COMPOSITIONS
20 REGIONS
800 RESONANCES
12 NUCLIDES
6. TYPICAL RUNNING TIME - RUNNING TIME IS DEPENDENT ON THE NUMBER OF REGIONS, NUMBER OF RESONANCES, AND NUMBER OF GROUPS. A PROBLEM WITH 25 RESONANCES AND 3000 FINE GROUPS REQUIRES 2 TO 4 MINUTES ON THE CDC3600.
7. UNUSUAL FEATURES OF THE PROGRAM - BECAUSE IT IS BASED ON SPACE AND LETHARGY DEPENDENT SLOWING DOWN SOURCES, RABBLE IS WELL-SUITED FOR PROBLEMS WHERE OVERLAPPING AND MUTUAL SHIELDING OF RESONANCES IS SIGNIFICANT. THE CELL MAY HAVE MORE THAN TWO COMPOSITIONS. EACH COMPOSITION MAY CONTAIN RESONANT NUCLIDES AND MAY BE AT A DIFFERENT TEMPERATURE.
8. RELATED AND AUXILIARY PROGRAMS - THIS PROGRAM IS AN EXTENSION OF RIFF RAFF (ACC ABSTRACT 213). IT USES THE QUICKW SUBROUTINE TO COMPUTE DOPPLER-BROADENED LINE SHAPE FUNCTIONS. THE PROGRAM WLIB IS USED TO PREPARE AN INPUT TAPE (LUN 10) CONTAINING INTERPOLATION TABLES FOR THE REAL AND IMAGINARY PARTS OF THE COMPLEX PROBABILITY INTEGRAL. THE PROGRAM FLAT IS USED TO PREPARE AN INPUT TAPE (LUN 11) CONTAINING THE INTERPOLATION TABLES FOR ESCAPE AND TRANSMISSION PROBABILITIES.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC3600 VERSION SUBMITTED JULY 1967, SAMPLE PROBLEM EXECUTED BY ACC.
10. REFERENCES - P. H. KIER AND A. A. ROBBA, RABBLE - A PROGRAM FOR COMPUTATION OF RESONANCE ABSORPTION IN MULTI-REGION REACTOR CELLS, ANL-7326, APRIL 1967.

10. REFERENCES (CONTINUED)

P. H. KIER, A METHOD OF COMPUTING RESONANCE INTEGRALS IN MULTI-REGION REACTOR CELLS, INTERNATIONAL CONFERENCE RESEARCH REACTOR UTILIZATION AND REACTOR MATHEMATICS, MEXICO CITY, MAY 1967.

ARGONNE CODE CENTER PROGRAMMING NOTE 68-7.

11. MACHINE REQUIREMENTS - 50K HIGH SPEED MEMORY WITH TWO INPUT TAPES
12. PROGRAMMING LANGUAGES USED - 3600 FORTRAN EXCEPT QUICKW WHICH IS IN ASSEMBLY LANGUAGE
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE MEMORY REQUIREMENTS CAN BE REDUCED TO PERMIT USE ON A 32K MACHINE BY REDUCING THE SIZE OF ARRAYS ASSOCIATED WITH THE NUMBER OF RESONANCES, THE NUMBER OF REGIONS OR COMPOSITIONS AND THE NUMBER OF GROUPS.
15. NAME AND ESTABLISHMENT OF AUTHOR -
P. H. KIER
REACTOR PHYSICS DIVISION
ARGONNE NATIONAL LABORATORY
9700 SOUTH CASS AVENUE
ARGONNE, ILLINOIS 60439
16. MATERIAL AVAILABLE -
SOURCE DECKS (RABBLE 847 CARDS, WLIB 209 CARDS, FLAT 139 CARDS)
BINARY DECKS (RABBLE 425 CARDS, WLIB 62 CARDS, FLAT 79 CARDS)
SAMPLE PROBLEM (RABBLE 39 CARDS)
REFERENCE REPORT, ANL-7326, AND NOTE
17. CATEGORY - B
KEYWORDS - CROSS SECTIONS, CELL CALCULATION, REACTION RATES, QUICKW CODES, RESONANCE ABSORPTION, FLUX AVERAGES

1. NAME OR DESIGNATION OF PROGRAM - SEALSHELL2 (M0110)
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE SEALSHELL2 PROGRAM DETERMINES STRESSES, STRAINS, DEFLECTIONS, AND REACTIONS IN A THICK SHELL OF REVOLUTION WITH AXISYMMETRIC LOADING. THE LOADING CONSISTS OF A TEMPERATURE DISTRIBUTION, INSIDE AND OUTSIDE PRESSURE DISTRIBUTIONS, AND CIRCUMFERENTIAL FORCES AND MOMENTS APPLIED TO THE MIDDLE SURFACE. THE SHELL IS LINEAR-ELASTIC WITH TENSILE, BENDING, AND SHEAR STRAINS.
4. METHOD OF SOLUTION - THE SHELL IS DIVIDED INTO 2 TO 100 SEGMENTS. EACH SEGMENT IS SUBJECTED TO 11 DEFORMATION SHAPES, WHOSE MAGNITUDES ARE CALCULATED BY THE PRINCIPLE OF VIRTUAL WORK. THAT IS, FOR SMALL VIRTUAL CHANGES IN MAGNITUDE, THE CHANGES IN STRAIN ENERGY AND EXTERNAL WORK ARE DETERMINED. THEN EQUATING THE ENERGY TO THE WORK GIVES A SET OF SIMULTANEOUS EQUATIONS WHICH MAY BE SOLVED FOR THE MAGNITUDES.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
100 SEGMENTS
11 DEFORMATION SHAPES PER SEGMENT
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM - THE PROGRAM WILL HANDLE THICK SHELLS OF REVOLUTION, AS WELL AS THE USUAL THIN SHELL STRUCTURES. AS DISCUSSED IN ADDENDUM I, AN OPTION ALLOWS THE REQUESTOR TO INPUT INFLUENCE COEFFICIENTS OF THE FLEXIBILITY MATRIX OF ANY SEGMENT. THE COEFFICIENTS MAY BE FOR ANY AXISYMMETRIC STRUCTURE, SUCH AS A TUBE SHEET IN A HEAT EXCHANGER. IN THIS WAY, DIFFERENT BRANCHES OF SHELLS MAY BE CONNECTED TOGETHER. ANOTHER APPLICATION IS TO CONDENSE A GROUP OF SEGMENTS INTO A SINGLE ELEMENT WHICH MAY BE USED IN A SERIES OF PROBLEMS AND SO CUT DOWN RUNNING TIME ON THE COMPUTER. ALSO, USING THE CONDENSED GROUPS OF SEGMENTS AS A SINGLE SEGMENT ALLOWS THE USER TO SPECIFY MORE ELEMENTS IN THE ANALYSIS OF A CRITICAL REGION IN THE SHELL.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED AUGUST 1967.
10. REFERENCES - C. M. FRIEDRICH, SEAL-SHELL-2, A COMPUTER PROGRAM FOR THE STRESS ANALYSIS OF A THICK SHELL OF REVOLUTION WITH AXISYMMETRIC PRESSURES, TEMPERATURES, AND DISTRIBUTED LOADS, WAPD-TM-398, DECEMBER 1963, AND ADDENDUM I, APRIL 1967.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.

11. MACHINE REQUIREMENTS - 32K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
SCOPE 2.C.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
AN OPTION IS INCLUDED TO PLOT THE SHELL GEOMETRY ON THE CDC280
MICROFILM RECORDER.
15. NAME AND ESTABLISHMENT OF AUTHOR -
C. M. FRIEDRICH
WESTINGHOUSE ELECTRIC CORPORATION
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P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122
16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (1767 CARDS)
SAMPLE PROBLEM (22 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS, WAPD-TM-398 ADDENDUM I, WAPD-TM-668
17. CATEGORY - I
KEYWORDS - STRESSES, SHELLS, PRESSURE DISTRIBUTION, TEMPERATURE
DISTRIBUTION, DEFORMATION, ELASTICITY

REACTOR CODE ABSTRACT 283

12/67

1. NAME OR DESIGNATION OF PROGRAM - M0552
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - M0552 SOLVES THE TRANSIENT RESPONSE PROBLEM OF LINEAR ELASTIC, LUMPED-MASS SYSTEMS SUBJECTED TO A UNIDIRECTIONAL FOUNDATION TRANSIENT THAT CAN BE EITHER A VELOCITY OR ACCELERATION TRANSIENT. NORMAL MODE THEORY IS USED AND THE INPUT TO THE PROGRAM CONSISTS OF THE MODE SHAPES, FREQUENCIES, AND FOUNDATION TRANSIENT. ELEMENT EFFECTS ARE ALSO EVALUATED AS A FUNCTION OF TIME. MODAL DAMPING COEFFICIENTS MAY BE SPECIFIED.
4. METHOD OF SOLUTION - THE FOUNDATION TRANSIENT IS SPECIFIED FOR EQUAL INCREMENTS OF TIME AND EITHER PARABOLIC OR LINEAR INTERPOLATION IS USED TO REPRESENT THE TRANSIENT FOR A GIVEN INTERVAL OF TIME. THIS FUNCTIONAL REPRESENTATION IS SUBSTITUTED INTO THE CONVOLUTION INTEGRAL AND A SUBSEQUENT EVALUATION IS MADE TO ESTABLISH THE TRANSIENT MODAL COORDINATES. THE REMAINDER OF THE SOLUTION FOLLOWS THE CLASSICAL NORMAL MODE SUPERPOSITION SCHEME. THE DIRECTION OF MOTION OF THE LUMPED MASSES IS RESTRICTED TO THE DIRECTION OF THE FOUNDATION TRANSIENT.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE TOTAL NUMBER OF POINTS USED TO DESCRIBE THE TRANSIENT IS LIMITED TO 2000. AN OPTION IS PROVIDED TO ENABLE SPECIFICATION OF A GENERAL TRIGONOMETRIC SERIES HAVING UP TO 20 TERMS, AS THE TRANSIENT. THE NUMBER OF MODES AND MASS POINTS THAT CAN BE CONSIDERED IS LIMITED TO 50.
6. TYPICAL RUNNING TIME - ABOUT 3 MINUTES ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED AUGUST 1967.
10. REFERENCES - E. A. ZANONI, M0552 A FORTRAN-4 PROGRAM FOR THE TIME DEPENDENT SOLUTION OF LINEAR ELASTIC SYSTEMS USING NORMAL MODE THEORY, WAPD-TM-657, MARCH 1967, AND ERRATA.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
11. MACHINE REQUIREMENTS - INPUT TAPE, OUTPUT TAPE, 2 SCRATCH TAPES, PRINTER, CDC-280 MICROFILM PLOTTER
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCCPE 2.0.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
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P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122
16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (634 CARDS)
SAMPLE PROBLEM (31 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS AND ERRATA
17. CATEGORY - I
KEYWORDS - STRUCTURAL ANALYSIS, STRESSES, DYNAMIC, TRANSIENT,
ELASTIC

REACTOR CODE ABSTRACT 284

12/67

1. NAME OR DESIGNATION OF PROGRAM - M0555(ACT1)
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - THIS PROGRAM WILL PREDICT THE PRESSURE TRANSIENT IN THE CONTAINMENT VESSEL AS A FUNCTION OF TIME AS A RESULT OF A MAJOR RUPTURE IN THE PRIMARY SYSTEM OF A PRESSURIZED WATER REACTOR. MEANS ARE AVAILABLE FOR INTRODUCING WATER AND HEAT INTO THE CONTAINMENT VESSEL. THERE ARE ALSO PROVISIONS FOR SPECIFYING VARIOUS TYPES OF HEAT SINKS FOR ENERGY ABSORPTION. THE TEMPERATURE DISTRIBUTION IN THESE HEAT SINKS IS CALCULATED AS A FUNCTION OF TIME. DISCHARGE RATES OF THE CONTAINMENT MAY ALSO BE DETERMINED AS A FUNCTION OF BOTH TIME AND THE INTERNAL PRESSURE.
4. METHOD OF SOLUTION - THE DIFFERENTIAL EQUATIONS ARE SOLVED BY A FORWARD DIFFERENCING TECHNIQUE.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THERE ARE THREE REGULAR INFLOW TABLES, ONE TABLE FOR SPRAY FLOW, 20 HEAT SINKS, TWO LINEAR AND ONE LOG-LOG HEAT INPUT TABLES AND TWO SPECIAL INFLOW TABLES FOR FLASH OUTPUT.
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - FLASH OUTPUT MAY BE USED DIRECTLY AS INPUT TO SPECIAL INPUT TABLES TO ACT1.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED AUGUST 1967.
10. REFERENCES - G. M. FULS, ACT-1 A DIGITAL PROGRAM FOR THE ANALYSIS OF THE CONTAINMENT TRANSIENT DURING A LOSS OF COOLANT ACCIDENT IN LWB, WAPD-TM-693, MAY 1967.
S. G. MARGOLIS AND J. A. REDFIELD, FLASH - A PROGRAM FOR DIGITAL SIMULATION OF THE LOSS-OF-COOLANT ACCIDENT, WAPD-TM-534, MAY 1966.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCCPE 2.0.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -

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WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION

MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (3277 CARDS)
SAMPLE PROBLEM (70 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS, WAPD-TM-668, WAPD-TM-693

17. CATEGORY - G

KEYWORDS - LWB REACTORS, CONTAINMENT, ACCIDENTS, WATER, HEATING,
PRESSURE DISTRIBUTION, TEMPERATURE DISTRIBUTION, COOL-
ANTS, STRESSES, LEAKAGE

REACTOR CODE ABSTRACT 285

12/67

1. NAME OR DESIGNATION OF PROGRAM - RESQ2/RESQ0/DBF1
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - RESQ2 CALCULATES THE RESONANCE INTEGRAL IN A TWO-DIMENSIONAL, HEXAGONAL SYSTEM CONSISTING OF FUEL, CLAD AND WATER WITH A REFLECTING BOUNDARY CONDITION.
4. METHOD OF SOLUTION - USING SUPERPOSITION, THE RESONANCE INTEGRAL IS DECOMPOSED INTO A SUM OF FIVE TERMS. THE FIRST TERM IS COMPUTED NUMERICALLY AND REPRESENTS A LARGE FRACTION OF THE TOTAL RESONANCE INTEGRAL. THE FOUR REMAINING TERMS ARE MONTE CARLO CORRECTIONS TO THE INTEGRAL. SINCE THE MONTE CARLO CORRECTION TERMS REPRESENT ONLY A SMALL CONTRIBUTION TO THE RESONANCE INTEGRAL, FEWER NEUTRON HISTORIES ARE REQUIRED TO ACHIEVE A SPECIFIC UNCERTAINTY THAN WOULD BE THE CASE IN AN ALL MONTE CARLO PROGRAM. IT IS ASSUMED THERE IS NO ABSORPTION IN WATER AND CLAD. ALL RESONANCE ISOTOPES ARE LUMPED INTO A SINGLE ISOTOPE HAVING AN AVERAGED MASS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
RESONANCE ISOTOPES ARE LUMPED TOGETHER WITH AN AVERAGED MASS.
MAXIMUM NUMBER OF ENERGY POINTS = 8001
MAXIMUM NUMBER OF ISOTOPES = 15
6. TYPICAL RUNNING TIME - 2.5 MINUTES ARE REQUIRED FOR 10,000 HISTORIES.
7. UNUSUAL FEATURES OF THE PROGRAM - RESQ2 IS MOST EFFICIENT WHEN THE ESCAPE PROBABILITY IS CLOSE TO ONE AND WHEN SCATTERING IS WEAK COMPARED TO RESONANCE ABSORPTION. THEREFORE, RESQ2 SHOULD BE USED MAINLY FOR THE STUDY OF RESONANCES IN THE RANGE FROM 0.625 TO 100 EV, WHERE RESONANCE SCATTERING IS NOT A PREDOMINANT EFFECT.
8. RELATED AND AUXILIARY PROGRAMS - RESQ0 PROVIDES A LIBRARY TAPE FOR RESQ2 PROBLEMS. DBF1 CALCULATES THE DOPPLER BROADENED $\Psi(\chi, \beta)$ AND $\Phi(\chi, \beta)$ FUNCTIONS.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED AUGUST 1967.
10. REFERENCES - B. L. ANDERSON, E. M. GELBARD, AND J. SPANIER, RESQ-2 A COMBINED ANALYTIC-MONTE CARLO CALCULATION OF RESONANCE ABSORPTION BASED ON SUPERPOSITION, WAPD-TM-665, JUNE 1967.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
11. MACHINE REQUIREMENTS - 40K MEMORY, ONE TAPE FOR THE LIBRARY TAPE, ONE TAPE OR DISK FOR SCRATCH
12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
SCOPE 2.C.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
IN ORDER TO CONVERT THIS PROGRAM TO ANOTHER COMPUTER OR OPERATING
SYSTEM, IT WOULD BE NECESSARY TO REWRITE THE INPUT ROUTINE AND THE
TAPE HANDLING ROUTINE.
15. NAME AND ESTABLISHMENT OF AUTHOR -
B. L. ANDERSON
WESTINGHOUSE ELECTRIC CORPORATION
BETTIS ATOMIC POWER LABORATORY
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122
16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECKS- (RESQ2 3895 CARDS, RESQO 860 CARDS,
DBF1 106 CARDS)
SAMPLE PROBLEMS (RESQ2 12 CARDS, RESQO 41 CARDS,
DBF1 9 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS
17. CATEGCRY - B
KEYWORDS - ESCAPE PROBABILITY, 2-DIMENSIONAL, CYLINDERS, HEXAGONAL
LATTICES, RESONANCE, DOPPLER BROADENING, MONTE CARLO
METHOD, BREIT-WIGNER FORMULA, SCATTERING, CAPTURE,
CROSS SECTIONS

REACTOR CODE ABSTRACT 286

12/67

1. NAME OR DESIGNATION OF PROGRAM - HOT2
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - HOT2 IS A DIGITAL COMPUTER PROGRAM TO SOLVE TWO-DIMENSIONAL PLANE AND AXIALLY SYMMETRIC STEADY-STATE AND TRANSIENT HEAT CONDUCTION PROBLEMS WITH DIAGONAL BOUNDARIES AND INTERFACES. MESH SPACING (AT MOST 5000 POINTS) IS COMPLETELY VARIABLE. AS MANY AS 99 REGIONS ARE PERMITTED IN ORDER TO DESCRIBE SPATIAL VARIATIONS IN MATERIAL PROPERTIES, HEAT GENERATION RATES, AND BOUNDARY CONDITIONS. THE HEAT GENERATION RATE AND BOUNDARY CONDITIONS MAY VARY WITH TIME.
4. METHOD OF SOLUTION - A NONUNIFORM MESH IS IMPOSED ON THE REGION OF SOLUTION, WITH MESH INTERVALS CHOSEN SO THAT ALL REGION INTERFACES PASS THROUGH AND CONNECT MESH POINTS IN EITHER A HORIZONTAL, VERTICAL, OR DIAGONAL MANNER. THE DIFFERENTIAL EQUATIONS ARE APPROXIMATED BY FIVE-POINT DIFFERENCE EQUATIONS AT THE MESH POINTS, AND THE RESULTING MATRIX EQUATION IS SOLVED BY THE ALTERNATING DIRECTION ITERATIVE TECHNIQUE.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - CONTACT RESISTANCES ARE NOT INCLUDED IN THE PROGRAM COMPUTATIONS, BUT MUST BE APPROXIMATED BY USE OF THIN REGIONS OF A LOW-CONDUCTIVITY MATERIAL. HEAT GENERATION RATES, BOUNDARY FILM COEFFICIENTS, AND BOUNDARY SINK TEMPERATURES ARE ASSUMED TO BE KNOWN. THE PROGRAM IS INTENDED TO HANDLE UP TO 99 REGIONS, 5000 MESH POINTS, 250 MESH POINTS IN EACH COORDINATE DIRECTION.
6. TYPICAL RUNNING TIME - A 1200 POINT PROBLEM WAS OBSERVED TO AVERAGE ABOUT .7 SECOND PER SWEEP (STEADY-STATE ITERATION OR TRANSIENT TIME-STEP) BUT THIS VARIED SOMEWHAT WITH DIFFERENT EDIT OPTIONS.
7. UNUSUAL FEATURES OF THE PROGRAM - HOT2 SOLVES THE TWO-DIMENSIONAL STEADY-STATE AND TRANSIENT HEAT CONDUCTION PROBLEM IN (R,Z) OR (X,Y) GEOMETRY WITH TIME-DEPENDENT HEAT GENERATION RATES, FILM COEFFICIENTS, AND SINK TEMPERATURES.
8. RELATED AND AUXILIARY PROGRAMS - ON REQUEST, FILES CONTAINING TEMPERATURE DISTRIBUTIONS ARE SAVED FOR SUBSEQUENT USE EITHER BY HOT2 OR OTHER PROGRAMS.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED SEPTEMBER 1967.
10. REFERENCES - W. D. PETERSON AND J. SPANIER, HOT-2 A TWO-DIMENSIONAL TRANSIENT HEAT CONDUCTION PROGRAM FOR THE CDC-6600, WAPD-TM-669, JUNE 1967.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.

11. MACHINE REQUIREMENTS - 64K MEMORY, AN ON-LINE CARD READER, PRINTER, AND PUNCH, ONE SYSTEM DISK AND 4 NON-SYSTEM DISKS, EACH ON ITS OWN CHANNEL
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 2.0.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE TERM OVERLAY IS ASSOCIATED WITH THE SCOPE 2.0 OPERATING SYSTEM. ESSENTIALLY AN OVERLAY DETERMINES THE AMOUNT OF PROGRAM TEXT IN CENTRAL MEMORY AT ANY MEMORY LOAD. WITH OVERLAY LOADING, THE MAIN OVERLAY IS ALWAYS IN MEMORY, AND, IN ADDITION, A PARTICULAR PRIMARY AND ONE SECONDARY OVERLAY MAY ALSO BE IN CENTRAL MEMORY. THE HOT2 PROGRAM CONTAINS A MAIN OVERLAY, FOURTEEN PRIMARY OVERLAYS, AND FIVE SECONDARY OVERLAYS. OVERALL PROGRAM CONTROL IS VESTED IN THE MAIN OVERLAY, AND THE PRIMARY OVERLAYS ARE LOADED AND EXECUTED THROUGH THE USE OF THE ROUTINE NEXT. WITH THE EXCEPTION OF ONE LABELED COMMON BLOCK HOT2, WHICH RESIDES WITH THE MAIN OVERLAY, ALL STORAGE ALLOCATION IS PERFORMED IN BLANK COMMON UNDER THE CONTROL OF FTB (REFERENCE 2). BLANK COMMON IS DECLARED IN THE MAIN OVERLAY AND IS GIVEN A LENGTH OF ONE. WHEN THIS OVERLAY IS LOADED, THE ACTUAL LENGTH OF BLANK COMMON IS OBTAINED FROM LABELED COMMON BLOCK SIZE AND PASSED TO FTB. ALL PRIMARY OVERLAYS ARE LOADED AT BLANK COMMON ORIGIN, AND AN FTB FILE (ID CALLED TEXT) IS RESERVED TO HOLD THESE OVERLAYS. THE FIVE SECONDARY OVERLAYS ARE LOADED UNDER THE CONTROL OF THEIR ASSOCIATED PRIMARY OVERLAYS FOR CERTAIN PROGRAM OPTIONS AND THESE SECONDARIES ARE DEFINED SO THAT A CERTAIN PORTION OF PROGRAM TEXT IS LOADED ONLY WHEN NEEDED. THE SUBROUTINE DUMMYO CONTAINED WITH THE MAIN OVERLAY IS NOT EXECUTED BUT IS USED TO CONTROL THE LOADING OF PROGRAM TEXT. DUMMYO FORCES THE LOADING OF THE FCHIP AND FTB ROUTINES INTO THE MAIN OVERLAY.
15. NAME AND ESTABLISHMENT OF AUTHOR -
W. D. PETERSON
WESTINGHOUSE ELECTRIC CORPORATION
BETTIS ATOMIC POWER LABORATORY
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122
16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (15,694 CARDS)
SAMPLE PROBLEM (39 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS
17. CATEGORY - H
KEYWORDS - 2-DIMENSIONAL, X-Y, R-Z, HEAT TRANSFER, TEMPERATURE DISTRIBUTION, SPACE-TIME

1. NAME OR DESIGNATION OF PROGRAM - BISYN
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - GE635
3. NATURE OF PHYSICAL PROBLEM SOLVED - BISYN SOLVES THE TWO-DIMENSIONAL MULTIGROUP NEUTRON DIFFUSION EQUATIONS IN X-Y OR R-Z GEOMETRY USING A NONITERATIVE SYNTHESIS METHOD. THIS APPROACH IS DESIGNED TO GREATLY REDUCE THE COMPUTER COST OF RUNNING TWO-DIMENSIONAL MULTIGROUP PROBLEMS AT THE RISK OF SOME LOSS IN ACCURACY OF THE DETAILED FLUX DISTRIBUTION.
4. METHOD OF SOLUTION - ONE OR TWO INITIAL ONE-DIMENSIONAL FLUX SOLUTIONS ARE RUN IN THE DIRECTION SPECIFIED BY THE USER. FROM THE INITIAL SOLUTION(S), GROUP AND REGION DEPENDENT BUCKLINGS ARE CALCULATED FOR UP TO 5 ONE-DIMENSIONAL FLUX AND ADJOINT SOLUTIONS IN THE ORTHOGONAL DIRECTION. THESE SOLUTIONS ARE NORMALIZED TO BECOME SHAPE FUNCTIONS. EACH SHAPE FUNCTION IS THEN USED TO CALCULATE FLUX AND ADJOINT WEIGHTED CROSS SECTIONS AVERAGED OVER THE FULL SPAN OF THAT SHAPE FUNCTION. THUS, REGION-DEPENDENT CROSS SECTIONS ARE OBTAINED WHICH VARY ONLY IN THE DIRECTION SELECTED FOR THE INITIAL SOLUTION(S). A FINAL SOLUTION IS NOW PERFORMED IN THIS DIRECTION TO GENERATE AN EIGENVALUE AND A MESH-DEPENDENT FLUX WHICH REPRESENTS THE VOLUME INTEGRATED FLUXES OVER THE ORTHOGONAL DIRECTION. THE SHAPE FUNCTIONS ARE THEN COMBINED WITH THE FINAL SOLUTION TO SYNTHESIZE A TWO-DIMENSIONAL SET OF FLUXES AND ADJOINTS. EXTENSIVE EDIT ROUTINES ARE AVAILABLE TO CALCULATE POWER DISTRIBUTION, NEUTRON BALANCE, ETC.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
 - MAXIMUM NUMBER OF REGIONS IN EACH DIMENSION = 8
 - MAXIMUM NUMBER OF MESH POINTS IN EACH DIMENSION = 75
 - MAXIMUM NUMBER OF ENERGY GROUPS = 16
 - MAXIMUM DOWNSCATTER = 8
6. TYPICAL RUNNING TIME - AN AVERAGE 16-GROUP PROBLEM TAKES ABOUT 3 MINUTES OF PROCESSOR TIME.
7. UNUSUAL FEATURES OF THE PROGRAM - SHAPE FUNCTIONS IN NON-MULTIPLYING REGIONS (BLANKETS, REFLECTORS) MAY BE CALCULATED USING A LEAKAGE SOURCE FROM THE SHAPE FUNCTION OF AN ADJACENT REGION, OR MAY BE SET UP SIMPLY BY COPYING A PREVIOUS SHAPE FUNCTION WITH THE FLUXES IN LETHARGY GROUPS $X+1$ TO $IMAX$ SET EQUAL TO THE FLUX IN GROUP X . BISYN CHECKS FOR NEGATIVE FLUX VALUES AND PRINTS OUT THE GROUP AND REGION WHERE THEY OCCUR. THE USER MAY SPECIFY THAT, IF NEGATIVE FLUXES OCCUR IN ANY SOLUTION, (A) THE PROGRAM TERMINATE, (B) SET THOSE BUCKLING VALUES TO ZERO WHICH ORIGINALLY GAVE A NEGATIVE TOTAL REMOVAL CROSS SECTION, OR (C) SET THOSE TOTAL REMOVAL CROSS SECTIONS TO ZERO WHICH WERE ORIGINALLY NEGATIVE, AND RE-RUN THAT SOLUTION.

8. RELATED AND AUXILIARY PROGRAMS - A FUEL CYCLE PROGRAM AND A PERTURBATION ROUTINE DESIGNED TO RUN WITH BISYN ARE UNDER PREPARATION.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
GE635 VERSION SUBMITTED JUNE 1967.
10. REFERENCES - P. GREEBLER, M. D. KELLEY, R. A. DAVIS, C. A. KECK, W. A. DUNCAN, BISYN - A TWO DIMENSIONAL SYNTHESIS PROGRAM, GEAP-4922, JULY 15, 1965.
MUG FILE TAPE DESCRIPTION, GE NOTE.
BISYN SUBROUTINE DESCRIPTIONS INCLUDING -
J. E. VAN ETEN, JR., ISUP, SEPTEMBER 1, 1966.
J. E. VAN ETEN, JR., FSUP, SEPTEMBER 1, 1966.
J. E. VAN ETEN, JR., GE-635 NPOST SPECIFICATION, APRIL 14, 1966.
FLEXO, J3T001, JANUARY 21, 1965.
NELS J. ANDERSON, JR., IDFTAP, A FORTRAN IV FUNCTION SUBPROGRAM, S6T001, JANUARY 15, 1965.
NELS J. ANDERSON, JR., RESTRT, Q3T001, AUGUST 1965.
SORT, L6T002.
NELS J. ANDERSON, JR., WRAPUP, A FORTRAN IV SUBROUTINE, Q3T001, AUGUST 1965.
BITA, L5T001, DECEMBER 3, 1964.
NELS J. ANDERSON, JR., FORTRAN IV SUBROUTINE ERRSYS, S4T001, JANUARY 15, 1965.
A. E. SCADDEN, ZERO, DIVISION OVERFLOW MODIFIER, Q4T001.
11. MACHINE REQUIREMENTS - 47,000 WORDS OF MEMORY, DISC, DRUM, 2 TAPES
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - GECOS (NED MODIFICATIONS).
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - AN INPUT LIBRARY IS REQUIRED. THE LIBRARY PROGRAM IS NOT INCLUDED.
15. NAME AND ESTABLISHMENT OF AUTHORS -
P. GREEBLER, M. D. KELLEY, R. A. DAVIS,
C. E. KECK, AND W. A. DUNCAN
GENERAL ELECTRIC COMPANY
NUCLEAR ENERGY DIVISION
ADVANCED PRODUCTS OPERATION
SUNNYVALE, CALIFORNIA
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (20,998 CARDS)
SAMPLE PROBLEM (123 CARDS)
LIBRARY (416 CARDS)
REFERENCE REPORT, NOTE, AND BISYN SUBROUTINE DESCRIPTIONS

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17. CATEGORY - C
KEYWORDS - 2-DIMENSIONAL, MULTIGROUP, DIFFUSION EQUATIONS, X-Y,
R-Z, FLUX SYNTHESIS, FLUX DISTRIBUTION, AVERAGES

1. NAME OR DESIGNATION OF PROGRAM - SNARG-1D
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC3600
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE PROGRAM IS DESIGNED TO SOLVE THE ONE-DIMENSIONAL NEUTRON TRANSPORT EQUATIONS. SNARG-1D IS WRITTEN FOR THE SOLUTION OF ONE-DIMENSIONAL PROBLEMS USING THE ORDER $N = 2, 4, 6, 8, 12, 16$, OR 32, AND APPLICABLE TO PLANE, CYLINDRICAL AND SPHERICAL GEOMETRIES. THE REAL OR ADJOINT SOLUTION MAY BE CALCULATED AND HOMOGENEOUS OR INHOMOGENEOUS PROBLEMS MAY BE SOLVED. FOUR CRITICALITY SEARCH OPTIONS ARE PROVIDED FOR WHICH A FIXED K_{eff} OR ALPHA, INVERSE PERIOD, VALUE MAY BE SPECIFIED RATHER THAN THE CRITICALITY VALUE, $K_{eff} = 1$. EITHER OF TWO INHOMOGENEOUS SOLUTIONS MAY BE OBTAINED (SHELL OR DISTRIBUTED SOURCE CALCULATIONS). LINEAR ANISOTROPIC COMPONENTS OF BOTH THE SHELL SOURCE AND THE SCATTERING CROSS SECTIONS MAY BE INCLUDED. FISSION SPECTRUM MATRICES ARE ALLOWED AS BOTH MATERIAL-DEPENDENT AND INCIDENT NEUTRON ENERGY-DEPENDENT FUNCTIONS.
4. METHOD OF SOLUTION - SNARG-1D IS BASED ON THE SN DISCRETE-ORDINATE METHOD AS DEVELOPED BY CARLSON. THE BOUNDARY CONDITIONS PROVIDED INCLUDE THE FREE, PERIODIC, REFLECTIVE, OR WHITE BOUNDARY. TO MAKE THE COMPUTATION OF ANGULAR FLUX STABLE WITH RESPECT TO ROUND-OFF ERRORS AND SPEED CONVERGENCE, THE DIRECTIONAL-EVALUATION RULE AND A SCALED ITERATION PROCEDURE ARE USED. IF THE USER IS NOT CONCERNED WITH ANGULAR FLUX VARIATION, A DIFFUSION THEORY OPTION IS INCLUDED IN THE CODE.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
6. TYPICAL RUNNING TIME - PROBLEM EXECUTION TIME VARIES FROM FIVE MINUTES TO SEVERAL HOURS DEPENDING ON THE PROBLEM SPECIFICATION.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - CROSS SECTION DATA ARE OBTAINED FROM TAPES GENERATED BY THE ANL CROSS SECTION LIBRARY PROGRAM, XLIBIT (ACC ABSTRACT 181).
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC3600 VERSION SUBMITTED MARCH 1968.
10. REFERENCES - G. J. DUFFY, H. GREENSPAN, S. D. SPARCK, J. V. ZAPATKA AND M. K. BUTLER, SNARG-1D, A ONE-DIMENSIONAL, DISCRETE-ORDINATE, TRANSPORT-THEORY PROGRAM FOR THE CDC-3600, ANL-7221, JUNE 1966.
S. D. SPARCK, XLIBIT ANL CROSS-SECTION LIBRARY CODE, ANL-7112, FEBRUARY 1966.
M. K. BUTLER AND H. GREENSPAN, GENERAL INPUT SPECIFICATIONS FOR ANL REACTOR PROGRAMS, ANL-7194, APRIL 1966.

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11. MACHINE REQUIREMENTS - A 2-BANK MACHINE IS REQUIRED.
12. PROGRAMMING LANGUAGE USED - 3600 FORTRAN
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
SCOPE (CDC360C).
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
G. J. DUFFY AND L. BRYANT
APPLIED MATHEMATICS DIVISION
ARGONNE NATIONAL LABORATORY
9700 SOUTH CASS AVENUE
ARGONNE, ILLINOIS 60439
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (16,747 CARDS)
SAMPLE PROBLEM (104 CARDS)
3600 LOAD AND GO OVERLAY TAPES (ON REQUEST ONLY)
REFERENCE REPORTS
17. CATEGORY - C
KEYWORDS - 1-DIMENSIONAL, MULTIGROUP, TRANSPORT THEORY, SN METHOD,
SLABS, CYLINDERS, SPHERES

1. NAME OR DESIGNATION OF PROGRAM - GAKER
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE GAKER CODE EVALUATES THE INELASTIC DOUBLE-DIFFERENTIAL NEUTRON SCATTERING CROSS SECTIONS FOR MODERATORS WITH PHONON SPECTRA WHICH CAN BE REPRESENTED AS SUMS OF DELTA-FUNCTIONS. IT IS BASED ON THE ORIGINAL MODEL FOR LIGHT WATER BY NELKIN, WHICH CONSISTED OF A TRANSLATOR, A HINDERED ROTATOR (TREATED AS AN ISOTROPIC OSCILLATOR), AND SEVERAL VIBRATIONAL OSCILLATORS. THE CODE HAS BEEN MODIFIED SEVERAL TIMES TO INCLUDE MORE OSCILLATORS AND TO TREAT ANISOTROPIC EFFECTS. FINAL ENERGY-INTEGRATED CROSS SECTIONS ARE ALSO CALCULATED.
4. METHOD OF SOLUTION -
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 75 ENERGIES
 - 30 SCATTERING ANGLES
 - 2 LEGENDRE MOMENTS OF THE SCATTERING KERNEL
 - 3 VIBRATIONAL OSCILLATORSONLY THE LATTER MAY NOT BE EASILY CHANGED.
6. TYPICAL RUNNING TIME - FOR 55 ENERGIES AND 15 ANGLES, 35 MINUTES ARE REQUIRED ON THE 7044, 15 MINUTES ON THE 7090, AND 6 MINUTES ON THE 1108.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
UNIVAC1108 VERSION SUBMITTED JULY 1967.
10. REFERENCES - M. S. NELKIN, THE SCATTERING OF SLOW NEUTRONS BY WATER, PHYSICAL REVIEW, 119, P. 741, 1960.
J. R. BEYSTER, J. R. BROWN, N. CORNGOLD, ET AL, INTEGRAL NEUTRON THERMALIZATION ANNUAL SUMMARY REPORT, GA-5798, OCTOBER 1964, PP. 155-162.
H. C. HONECK, THERMOS A THERMALIZATION TRANSPORT THEORY CODE FOR REACTOR LATTICE CALCULATIONS, BNL-5826, SEPTEMBER 1961.
INPUT FOR ANISOTROPIC VERSION OF GAKER, GA NOTE.
11. MACHINE REQUIREMENTS - 32K MEMORY AND 2 SCRATCH FILES
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - UNIVAC1108 EXEC II.

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14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
D. H. HOUSTON
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112
16. MATERIAL AVAILABLE -
SOURCE DECK (1108-390 CARDS)
SAMPLE PROBLEM (1108-15 CARDS)
REFERENCE REPORT, GA-5798 PP. 155-162, AND NOTE
17. CATEGORY - A
KEYWORDS - INELASTIC SCATTERING CROSS SECTIONS, MODERATORS, ANISOTROPIC SCATTERING, LIGHT WATER, THERMAL, JAG CODES, DTOK CODES, GAD CODES, GANG CODES, GAIN CODES

1. NAME OR DESIGNATION OF PROGRAM - GASA (GENERAL ATOMIC STABILITY ANALYSIS)
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108.
3. NATURE OF PHYSICAL PROBLEM SOLVED - GASA DETERMINES THE STABILITY OF ANY PHYSICAL SYSTEM WHOSE MOTION IS DESCRIBABLE BY A SET OF FIRST-ORDER LINEAR DIFFERENTIAL EQUATIONS. IN PARTICULAR GASA EVALUATES THE STABILITY OF A REACTOR DESCRIBED BY THE LINEAR REACTOR KINETICS EQUATIONS WITH TEMPERATURE FEEDBACK, AGAINST PERTURBATIONS ABOUT ANY OPERATING POWER LEVEL. THE PROGRAM WILL ALSO CALCULATE AND PLOT THE TRANSFER FUNCTION BETWEEN ANY STATE VARIABLE OF THE SYSTEM AND A GIVEN EXTERNAL FORCING FUNCTION (SUCH AS AN EXTERNAL REACTIVITY PERTURBATION) OR ANOTHER STATE VARIABLE OF THE SYSTEM.
4. METHOD OF SOLUTION - GASA DETERMINES THE ROOTS OF A SET OF FIRST-ORDER LINEAR DIFFERENTIAL EQUATIONS, ACCORDING TO THE LYAPUNOV FIRST APPROXIMATION THEOREMS, THEREBY DETERMINING THE STABILITY OF THE SYSTEM.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE MAXIMUM NUMBER OF EQUATIONS THAT CAN BE HANDLED IS 25.
6. TYPICAL RUNNING TIME - 6 SECONDS ARE REQUIRED ON THE IBM7044 FOR AN 8 X 8 MATRIX (I.E. 8 EQUATIONS IN THE SET).
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
UNIVAC1108 VERSION SUBMITTED JULY 1967.
10. REFERENCES - A. M. BAXTER, GASA - A GENERAL LINEAR STABILITY ANALYSIS CODE, GAMC-7242, JULY 15, 1966.
A. M. LETOV, STABILITY IN NON-LINEAR CONTROL SYSTEMS, CHAPTER 1, 1961.
R. BELLMAN, STABILITY THEORY OF DIFFERENTIAL EQUATIONS, CHAPTER 1, 1953.
F. P. IMAD AND J. E. VAN NESS, EIGENVALUES BY THE QR TRANSFORM, SDP/3-CATALOG OF 704/704C/7044/7090/7094 PROGRAMS DISTRIBUTED BY SHARE (GENERAL DYNAMICS/CONVAIR), PAGE 48.
E. P. GYFTOPOLOUS, THE TECHNOLOGY OF NUCLEAR REACTOR SAFETY, VOL. I, CHAPTER 3, 1964.
11. MACHINE REQUIREMENTS - 64K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV

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13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
EXEC II (UNIVAC1108).
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
A. BAXTER
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112
16. MATERIAL AVAILABLE -
SOURCE DECK (1108-1011 CARDS)
SAMPLE PROBLEM (1108-28 CARDS)
REFERENCE REPORT, GAMC-7242
17. CATEGORY - E
KEYWORDS - SPACE-INDEPENDENT, REACTOR KINETICS, TEMPERATURE FEED-
BACK, TRANSFER-FUNCTION, GRAPHS

1. NAME OR DESIGNATION OF PROGRAM - HEXSCAT
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - HEXSCAT CALCULATES P0 THROUGH P3 COMPONENTS OF THE POLYCRYSTALLINE COHERENT ELASTIC NEUTRON SCATTERING CROSS SECTION PER NUCLEUS FOR A HEXAGONAL LATTICE. THE CODE AVERAGES POINT VALUES OVER INPUT GROUP BOUNDARIES TO GIVE SMOOTHED GROUP CROSS SECTIONS.
4. METHOD OF SOLUTION -
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
101 ENERGY GROUPS
24 LEGENDRE MOMENTS
6. TYPICAL RUNNING TIME - ABOUT 1 MINUTE IS REQUIRED FOR LOW ENERGIES, LONGER FOR HIGH ENERGIES.
7. UNUSUAL FEATURES OF THE PROGRAM - THE CROSS SECTION BECOMES SMOOTH AT HIGH ENERGIES AND IS USUALLY OBTAINED BY INTERPOLATING WITHIN A SMALL TABLE OF VALUES, RATHER THAN CALCULATING DIRECTLY ALL NEEDED VALUES.
8. RELATED AND AUXILIARY PROGRAMS - A MODIFIED VERSION OF THE CODE HAS BEEN WRITTEN TO SUPPLY DATA IN A GENERAL (NEARLY ENERGY-MESH-INDEPENDENT) FORM FOR THE EVALUATED NUCLEAR DATA FILE (ENDF). THIS VERSION CALCULATES THE P0 THROUGH P5 MOMENTS AND PROVIDES FOR A LARGER NUMBER OF POINT VALUES OF THE CROSS SECTIONS. NO GROUP AVERAGING IS DONE IN THIS VERSION. THE POINT VALUES ARE TO BE AVERAGED LATER BY THE FLANGE2 CODE AT THE SAME TIME THE INELASTIC DATA ARE TREATED.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
UNIVAC1108 VERSION SUBMITTED JULY 1967.
10. REFERENCE - Y. D. NALIBOFF AND J. U. KOPPEL, HEXSCAT COHERENT ELASTIC SCATTERING OF NEUTRONS BY HEXAGONAL LATTICES, GA-6026, DECEMBER 15, 1964.
11. MACHINE REQUIREMENTS - 64K UNIVAC1108
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II (UNIVAC1108).
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

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15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE -

SOURCE DECK (1108-358 CARDS)
REFERENCE REPORT

17. CATEGORY - B

KEYWORDS - ELASTIC SCATTERING CROSS SECTIONS, HEXAGONAL LATTICES,
ANISOTROPIC SCATTERING, AVERAGES

1. NAME OR DESIGNATION OF PROGRAM - PSEUDO
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - RESONANCE PARAMETERS ARE CONSTRUCTED FROM AVERAGE NUCLEAR PROPERTIES IN THE RESOLVED RESONANCE REGION.
4. METHOD OF SOLUTION - RANDOM SAMPLING TECHNIQUES ARE USED TO SELECT FROM ASSUMED DISTRIBUTION FUNCTIONS FOR LEVEL SPACINGS AND WIDTHS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
500 LADDERS
4000 RESONANCES PER LADDER
6. TYPICAL RUNNING TIME - 9 SECONDS ARE REQUIRED FOR ONE LADDER OF 1248 RESONANCES.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - OUTPUT DATA MAY BE PREPARED FOR THE GAM2, GARCL (ACC ABSTRACT 219), OR GAR (ACC ABSTRACT 316) PROGRAMS.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
UNIVAC1108 VERSION SUBMITTED JULY 1967.
10. REFERENCE - M. W. DYOS, CONSTRUCTION OF STATISTICAL RESONANCES IN THE UNRESOLVED RESONANCE REGION, THE PSEUDO CODE, GA-7136, MAY 26, 1966.
11. MACHINE REQUIREMENTS - 15K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II (UNIVAC1108).
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE USER CAN CHOOSE EITHER TWO OR THREE DEGREES OF FREEDOM FOR THE DISTRIBUTION OF FISSION WIDTHS.
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE -

SOURCE DECK (326 CARDS)
SAMPLE PROBLEM (6 CARDS)
REFERENCE REPORT

17. CATEGORY - A

KEYWORDS - RESOLVED REGION, RESONANCE PARAMETERS, STATISTICS,
INPUT DATA PREPARATION

1. NAME OR DESIGNATION OF PROGRAM - MARS
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7094
3. NATURE OF PHYSICAL PROBLEM SOLVED - MARS PERFORMS A COUPLED NEUTRONICS-HYDRODYNAMICS CALCULATION FOR A FINITE CYLINDER CORE WITH CONCENTRIC REGIONS OF DIFFERENT COMPOSITIONS AND CHARACTERISTICS TO OBTAIN THE TIME-DEPENDENT ENERGY RELEASE THAT RESULTS FROM THE INSERTION OF REACTIVITY ACCORDING TO A PRESCRIBED PROGRAM. THE BASIC USE FOR THIS PROGRAM IS FOUND IN ANALYSIS OF FAST REACTOR CORE COLLAPSE ACCIDENTS. THE REACTIVITY CHANGES DUE TO MATERIAL DISPLACEMENT IS DETERMINED BY THE USE OF PERTURBATION THEORY AND THE PRESSURE GENERATION FOR THE HYDRODYNAMICS CALCULATION IS DETERMINED BY THE THE USE OF AN EXPONENTIAL FORM FOR THE SATURATED VAPOR PRESSURE CURVE. THE DOPPLER EFFECT IS DETERMINED BY THE USE OF A FLEXIBLE $T^{*}(-N)$ FORM WHERE N IS A PARAMETER THAT IS SPECIFIED AT THE TIME OF EXECUTION. THE EXTERNAL REACTIVITY INSERTION CAN BE IN THE FORM OF A STEP, A LINEAR RAMP OR A PARABOLIC RAMP, WITH A LIMIT ON THE AMOUNT OF REACTIVITY WHICH CAN BE INSERTED BEING A DEFINABLE QUANTITY. TABULATED VALUES OF THE POWER AND MATERIAL WORTH DISTRIBUTIONS ARE ACCEPTED AS INPUT. THE CHARACTERISTICS OF EACH REGION (DENSITY, SPECIFIC HEAT, DOPPLER COEFFICIENT, EQUATION OF STATE, GEOMETRIC LIMITS) ARE SPECIFIED INDEPENDENTLY FOR EACH REGION TO PROVIDE FLEXIBILITY IN THE TREATMENT OF CORES OF UNUSUAL CONFIGURATION. THE NEUTRON KINETICS CALCULATIONS ARE DONE WITH A POINT REACTOR MODEL AND DELAYED NEUTRONS ARE USED.
4. METHOD OF SOLUTION - THE APPROXIMATE HYDRODYNAMICS EQUATIONS AND THE NEUTRON KINETICS EQUATIONS ARE SOLVED BY THE RUNGE-KUTTA METHOD. THE FEEDBACK REACTIVITY DUE TO MATERIAL MOTION IS CALCULATED BY SIMPLE NUMERICAL SUMMATIONS TO APPROXIMATE SURFACE AND VOLUME INTEGRALS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
6 REGIONS
6 DELAYED NEUTRON GROUPS
6. TYPICAL RUNNING TIME - 5 MINUTES ARE REQUIRED WHEN 2 REGIONS ARE USED AND THE PROBLEM IS STARTED FROM DELAYED CRITICAL.
7. UNUSUAL FEATURES OF THE PROGRAM - MARS WILL SOLVE THE SEVERE EXCURSION PROBLEM IN AN R-Z GEOMETRY FOR A CYLINDRICAL CORE WITH AS MANY AS 6 REGIONS.
8. RELATED AND AUXILIARY PROGRAMS - THE RUNGE-KUTTA SUBROUTINE SPDEQ (GENERAL MOTORS) IS USED.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
IBM7094 VERSION SUBMITTED AUGUST 1967.

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10. REFERENCES - N. HIRAKAWA, MARS, A TWO-DIMENSIONAL EXCURSION CODE, APDA-198, JUNE 1967.
DIFFERENTIAL EQUATION SUBROUTINE - SPDEQ, APDA NCTE.
11. MACHINE REQUIREMENTS - 32K IBM7094
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - IBSYS, VERSION 13.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
N. HIRAKAWA AND A. E. KLINKMAN
ATOMIC POWER DEVELOPMENT ASSOCIATES, INCORPORATED
1911 FIRST STREET
DETROIT, MICHIGAN 48226
16. MATERIAL AVAILABLE -
SOURCE DECK (701 CARDS)
SAMPLE PROBLEM (57 CARDS)
REFERENCE REPORT AND NOTE
17. CATEGORY - F
KEYWORDS - 2-DIMENSIONAL, EXCURSIONS, R-Z GEOMETRIES, FAST REACTORS, SPDEQ CODES, SPACE-INDEPENDENT KINETICS, REACTIVITY FEEDBACK, HYDRODYNAMICS, EXPANSION

1. NAME OR DESIGNATION OF PROGRAM - M0899/HOH
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - BY MAKING CALLS ON A SUBROUTINE CALLED HOH, M0899 EDITS THERMODYNAMIC AND TRANSPORT PROPERTIES OF WATER OVER THE RANGE 14.5 TO 2538 PSIA AND UP TO 608 DEGREES FAHRENHEIT BELOW SATURATION AND 932 DEGREES FAHRENHEIT ABOVE SATURATION.
4. METHOD OF SOLUTION - THE THERMODYNAMIC DATA IS STORED AS TABLES AND VALUES ARE OBTAINED BY BILINEAR INTERPOLATION IN PRESSURE AND TEMPERATURE. TRANSPORT PROPERTIES ARE COMPUTED FROM CURVE FITS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
6. TYPICAL RUNNING TIME - LESS THAN ONE MINUTE IS REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - M0899 IS AN EXTENSION OF SUBROUTINE HOH. IT MAKES CALLS ON HOH TO GET PROPERTIES FOR PRESSURES AND TEMPERATURES SUPPLIED AND THEN EDITS THE PROPERTIES. HOH CAN BE USED APART FROM M0899 BY ANY OTHER FORTRAN IV PROGRAM TO OBTAIN WATER PROPERTIES.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED AUGUST 1967.
10. REFERENCES - L. L. LYNN, A DIGITAL COMPUTER PROGRAM FOR NUCLEAR REACTOR DESIGN WATER PROPERTIES, WAPD-TM-680, JULY 1967.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
R. W. BAIN, NATIONAL ENGINEERING LABORATORY STEAM TABLES, HER MAJESTY'S STATIONERY OFFICE, EDINBURGH, 1964.
11. MACHINE REQUIREMENTS - A FORTRAN IV COMPILER PLUS ABOUT 3000 LOCATIONS FOR SUBROUTINE HOH AND 10,000 LOCATIONS FOR THE MAIN PROGRAM.
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 2.C.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

REACTOR CODE ABSTRACT 294

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15. NAME AND ESTABLISHMENT OF AUTHOR -

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BETTIS ATOMIC POWER LABORATORY
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WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION

MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (1022 CARDS)
SAMPLE PROBLEM (7 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS, WAPD-TM-680 AND WAPD-TM-668

17. CATEGORY - F

KEYWORDS - THERMODYNAMICS, WATER, PRESSURE, TEMPERATURE, LIQUIDS,
VAPORS

1. NAME OR DESIGNATION OF PROGRAM - FLASH2
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - FLASH2 DETERMINES THE TRANSIENT RESPONSE OF A WATER REACTOR TO A LOSS-OF-COOLANT ACCIDENT OR SEVERE VARIABLE PRESSURE OPERATION.
4. METHOD OF SOLUTION - THE NUMERICAL INTEGRATIONS ARE DONE IN A SERIAL MANNER BY A COMBINATION OF FORWARD DIFFERENCING AND IMPLICIT TECHNIQUES. FLUID PROPERTIES ARE OBTAINED FROM PROGRAM TABLES.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE REACTOR PLANT MUST BE REPRESENTED BY 20 OR LESS PRESSURE-DETERMINING NODES AND 40 OR LESS FLOW PATHS. THE PLANT AND CORE GEOMETRY ARE ASSUMED FIXED IN TIME. HENCE, THE SOLUTION IS NOT APPLICABLE AFTER SIGNIFICANT CORE MELTING.
6. TYPICAL RUNNING TIME - RUNNING TIME VARIES BETWEEN 1 TO 100 TIMES REAL TIME, DEPENDING ON THE COMPLEXITY OF THE PROBLEM.
7. UNUSUAL FEATURES OF THE PROGRAM - FLASH2 CAN BE APPLIED TO ANY WATER REACTOR GEOMETRY OR PLANT CONFIGURATION. IT INCLUDES ACOUSTIC EFFECTS FOR BOTH SUBCOOLED AND SATURATED BLOWDOWNS.
8. RELATED AND AUXILIARY PROGRAMS - FLASH2 IS AN EXTENSION OF FLASH. FLASH2 USES THE INPUT PROCESSING SUBROUTINES FCHIP, CARDS, AND INPF (SEE REFERENCE 3).
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED SEPTEMBER 1967.
10. REFERENCES - J. A. REDFIELD, J. H. MURPHY, AND V. C. DAVIS, FLASH-2 A FORTRAN IV PROGRAM FOR THE DIGITAL SIMULATION OF A MULTINODE REACTOR PLANT DURING LOSS OF COOLANT, WAPD-TM-666, APRIL 1967.
S. G. MARGOLIS AND J. A. REDFIELD, FLASH A PROGRAM FOR DIGITAL SIMULATION OF THE LOSS-OF-COOLANT ACCIDENT, WAPD-TM-534, MAY 1966.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
11. MACHINE REQUIREMENTS - 64K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 2.0.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

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12/67

15. NAME AND ESTABLISHMENT OF AUTHORS -

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WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION

MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (5640 CARDS)
SAMPLE PROBLEM (46 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS, WAPD-TM-666 AND WAPD-TM-668

17. CATEGORY - G

KEYWORDS - ACCIDENT, CONTAINMENT, WATER REACTORS, FUELS, COOLANTS,
FLUID FLOW, PRESSURE DISTRIBUTION, TEMPERATURE DISTRIBUTION

1. NAME OR DESIGNATION OF PROGRAM - GRDWRK
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - GRDWRK GENERATES AS PUNCHED OUTPUT THE BASIC FINITE ELEMENT REFERENCE GRID WORK FOR THE SAFE CODES. THIS GENERATED GRID CONSISTS OF TRIANGULAR ELEMENTS AND NODES, UNIAXIAL ELEMENTS, SUCH AS REINFORCEMENT BARS, TENDONS, AND ANCHORS, AND BIAXIAL MEMBRANES, SUCH AS ANY THIN SHELL OR LINER. THE PUNCHED OUTPUT SERVE AS DIRECT INPUT DATA TO THE SAFE CODES.
4. METHOD OF SOLUTION - BY PUNCHED CARD INPUT, THE USER DESCRIBES THE STRUCTURE, PRESSURE LOADING, AND TYPE OF GRID DESIRED. THE CODE SOLVES SIMPLE SIMULTANEOUS LINEAR EQUATIONS TO GENERATE THE COORDINATES OF EACH GRID NODE, NUMBERS ALL NODES AND ELEMENTS BY A SYSTEM OF INDEXING, COMPUTES EXTERNAL PRESSURE LOADS ON SPECIFIED NODES BY ASSUMING AXISYMMETRICAL DISTRIBUTION, TRACES GIVEN UNIAXIAL AND BIAXIAL ELEMENTS ACROSS THE BACKGROUND GRID TO ESTABLISH APPROXIMATIONS TO THESE ELEMENTS WITHIN THE GRID, AND PUNCHES THE RESULTS OUT AS DATA CARDS TO BE USED WITH A SAFE CODE.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
1200 GRID NODES
3600 TOTAL GRID, UNIAXIAL, AND BIAXIAL ELEMENT NODES
6. TYPICAL RUNNING TIME - 1 TO 5 SECONDS ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM - GRDWRK GENERATES THE UNIAXIAL AND BIAXIAL APPROXIMATIONS.
8. RELATED AND AUXILIARY PROGRAMS - SAFE-PCRS (ACC ABSTRACT 250), SAFE-AXISYM (ACC ABSTRACT 251), SAFE-PLANE (ACC ABSTRACT 252), SAFE-SHELL (ACC ABSTRACT 253), AND SAFE-CREEP (ACC ABSTRACT 300)
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
UNIVAC1108 VERSION SUBMITTED AUGUST 1967.
10. REFERENCES - D. A. DOERING, GRDWRK, A COMPUTER PROGRAM FOR THE GENERATION OF TWO-DIMENSIONAL FINITE ELEMENT GRIDS, A USERS MANUAL, GA-7856, JUNE 30, 1967.
D. C. CORNELL, SAFE-PCRS, A COMPUTER PROGRAM FOR THE STRESS ANALYSIS OF COMPOSITE BODIES OF REVOLUTION, INPUT INSTRUCTIONS, GA-6588, AUGUST 1, 1965.
D. C. CORNELL, SAFE-PLANE, A COMPUTER PROGRAM FOR THE STRESS ANALYSIS AND DESIGN OF TWO-DIMENSIONAL COMPOSITE BODIES, A USERS MANUAL, GA-7851, JUNE 30, 1967.
D. C. CORNELL, SAFE-SHELL, A COMPUTER PROGRAM FOR THE STRESS ANALYSIS OF THIN SHELL BODIES OF REVOLUTION, A USERS MANUAL, GA-7852, JUNE 30, 1967.

10. REFERENCES (CONTINUED)

Y. RASHID AND F. J. DE ARRIAGA, SAFE-CREEP, COMPUTER PROGRAM FOR THE VISCOELASTIC ANALYSIS OF AXISYMMETRIC AND PLANE CONCRETE STRUCTURES, A USERS MANUAL, GA-7854, JULY 31, 1967.

Y. RASHID, C. C. CORNELL, AND K. B. JADHAV, SAFE-3D, A COMPUTER PROGRAM FOR THE THREE-DIMENSIONAL STRESS ANALYSIS OF COMPOSITE STRUCTURES - A USERS MANUAL, SEPTEMBER 1, 1967.

11. MACHINE REQUIREMENTS - 20,100 WORDS PLUS OPERATING SYSTEM

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - UNIVAC EXEC II, GAX23B THROUGH GAX27.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE PROGRAM IS OPERABLE ON ANY COMPUTER WITH FORTRAN IV AND 25K CORE.

15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE -

SOURCE DECK (1339 CARDS)
SAMPLE PROBLEM (73 CARDS)
REFERENCE REPORT, GA-7856

17. CATEGORY - L

KEYWORDS - INPUT DATA PREPARATION, 2-DIMENSIONAL GEOMETRIES, PARAMETERS, SAFE CODES, STRESSES

1. NAME OR DESIGNATION OF PROGRAM - CONTEMPT
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7040,44
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE CONTEMPT COMPUTER PROGRAM PREDICTS THE PRESSURE-TEMPERATURE RESPONSE OF A DRY WELL CONTAINMENT BUILDING TO A LOSS-OF-COOLANT ACCIDENT. THE CONTAINMENT VOLUME IS SEPARATED INTO A LIQUID REGION AND A VAPOR REGION. EACH REGION IS ASSUMED TO HAVE A UNIFORM TEMPERATURE BUT THE TEMPERATURE OF THE TWO REGIONS MAY BE DIFFERENT. THE CONTAINMENT BUILDING IS REPRESENTED AS CONSISTING OF SEVERAL HEAT CONDUCTING STRUCTURES WHOSE THERMAL BEHAVIOR CAN BE DESCRIBED BY THE ONE-DIMENSIONAL MULTIREGION HEAT CONDUCTION EQUATION. WATER AND ENERGY RATES FROM DISCHARGE OF COOLANT, THE BOILING OF RESIDUAL WATER BY REACTOR DECAY HEAT, THE SUPERHEATING OF STEAM PASSING THROUGH THE CORE, AND METAL-WATER REACTIONS ARE ASSUMED AVAILABLE FROM PREVIOUS CALCULATIONS AND ARE INPUT DATA TO CONTEMPT. PROGRAM OUTPUT INCLUDES CONTAINMENT VOLUME PRESSURE AND TEMPERATURES, TEMPERATURE THROUGH BUILDING STRUCTURES, AND THE AMOUNT OF WATER, VAPOR, AND ENERGY IN THE CONTAINMENT VOLUMES. THE PROGRAM ALSO CALCULATES BUILDING LEAKAGE AND THE EFFECTS OF ENGINEERED SAFEGUARD DEVICES SUCH AS WATER SPRAY AND FAN COOLER SYSTEMS.
4. METHOD OF SOLUTION - THE INITIAL CONDITIONS OF THE CONTAINMENT ATMOSPHERE ARE CALCULATED FROM INPUT VALUES AND THE INITIAL TEMPERATURE DISTRIBUTION THROUGH THE CONTAINMENT STRUCTURES ARE DETERMINED FROM THE STEADY-STATE SOLUTION OF THE HEAT CONDUCTION EQUATIONS. A TIME ADVANCEMENT PROCEEDS AS FOLLOWS. THE INPUT WATER AND ENERGY RATES ARE EVALUATED AT THE MIDPOINT OF A TIME INTERVAL AND ADDED TO THE CONTAINMENT SYSTEM. LEAKAGE AND ENGINEERED SAFEGUARD EFFECTS ARE CALCULATED USING AN EXPLICIT ADVANCEMENT PROCEDURE. HEAT LOSSES OR GAINS, APPROPRIATE FOR THE BEGINNING OF THE TIME-STEP, ARE ADDED TO THE CONTAINMENT SYSTEM. CONTAINMENT VOLUME PRESSURE AND TEMPERATURES ARE ESTIMATED BY SOLVING THE MASS, VOLUME, AND ENERGY BALANCE EQUATIONS. USING THESE RESULTS AS BOUNDARY CONDITIONS, THE HEAT CONDUCTION EQUATIONS DESCRIBING STRUCTURE BEHAVIOR ARE ADVANCED USING AN IMPLICIT TECHNIQUE. THE RESULTING HEAT TRANSFER RATES ARE USED TO CORRECT THE PREVIOUS ESTIMATES OF THE WATER AND ENERGY STORAGE IN THE CONTAINMENT VOLUME AND THE CONTAINMENT CONDITIONS ARE OBTAINED BY SOLVING FOR THE SECOND TIME THE CONTAINMENT BALANCE EQUATIONS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
20 HEAT STRUCTURES
101 MESH POINTS PER STRUCTURE
SEVERAL METHODS FOR DESCRIBING NON-CONSTANT HEAT TRANSFER COEFFICIENTS ARE PROVIDED, BUT THE THERMAL CONDUCTIVITY AND HEAT CAPACITY ARE CONSTANT. WATER THERMODYNAMIC PROPERTIES ARE DETERMINED FROM TABLE LOOKUP AND INTERPOLATION PROCEDURES.

6. TYPICAL RUNNING TIME - ON THE IBM7040 APPROXIMATELY 0.32 SECOND PER TIME ADVANCEMENT IS REQUIRED FOR A PROBLEM USING 90 MESH POINTS FOR HEAT STRUCTURES.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
IBM7040 VERSION SUBMITTED SEPTEMBER 1967.
10. REFERENCE - L. C. RICHARDSON, L. J. FINNEGAN, R. J. WAGNER, AND J. M. WAAGE, CONTEMP, A COMPUTER PROGRAM FOR PREDICTING THE CONTAINMENT PRESSURE-TEMPERATURE RESPONSE TO A LOSS-OF-COOLANT ACCIDENT, ICC-17220, JUNE 1967.
11. MACHINE REQUIREMENTS - 32K MEMORY
12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND MAP (DECIMAL INPUT, DATE, AND CLOCKING ROUTINES)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - IBSYS.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
L. C. RICHARDSON, L. J. FINNEGAN, R. J. WAGNER,
AND J. M. WAAGE
ATOMIC ENERGY DIVISION
PHILLIPS PETROLEUM COMPANY
P. O. BOX 2067
IDAHO FALLS, IDAHO 83401
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (7040-2154 CARDS)
SAMPLE PROBLEM (7040-58 CARDS)
REFERENCE REPORT
17. CATEGORY - G
KEYWORDS - ACCIDENTS, TEMPERATURE DISTRIBUTION, PRESSURE DISTRIBUTION, CONTAINMENT, COOLANTS, WATER, LIQUIDS, VAPORS, LEAKAGE, HEAT TRANSFER

1. NAME OR DESIGNATION OF PROGRAM - GGC4
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE GGC4 PROGRAM SOLVES THE MULTIGROUP SPECTRUM EQUATIONS WITH SPATIAL DEPENDENCE REPRESENTED BY A SINGLE POSITIVE INPUT BUCKLING. BROAD GROUP CROSS SECTIONS (SHIELDED OR UNSHIELDED) ARE PREPARED FOR DIFFUSION AND TRANSPORT CODES BY AVERAGING WITH THE CALCULATED SPECTRA OVER INPUT-DESIGNATED ENERGY LIMITS. THE CODE IS DIVIDED INTO THREE MAIN PARTS. A FAST (GAM) SECTION WHICH COVERS THE ENERGY RANGE FROM 14.9 MEV TO 0.414 EV, A THERMAL (GATHER) SECTION WHICH COVERS THE ENERGY RANGE FROM 0 TO 2.38 EV, AND A COMBINING (COMBO) SECTION WHICH COMBINES FAST AND THERMAL CROSS SECTIONS INTO SINGLE SETS. BASIC NUCLEAR DATA FOR THE FAST SECTION WHICH CONSISTS OF FINE GROUP-AVERAGED CROSS SECTIONS AND RESONANCE PARAMETERS IS READ OFF A DATA TAPE. THE FINE GROUP ABSORPTION AND FISSION CROSS SECTIONS MAY BE ADJUSTED BY PERFORMING A RESONANCE INTEGRAL CALCULATION. UTILIZING A FISSION SOURCE AND AN INPUT BUCKLING, THE CODE SOLVES THE P1, B1, B2, OR B3 APPROXIMATION TO OBTAIN THE ENERGY-DEPENDENT FAST SPECTRUM. TWO OR SIX SPATIAL MOMENTS OF THE SPECTRUM (DUE TO A PLANE SOURCE) MAY ALSO BE EVALUATED. INSTEAD OF PERFORMING A SPECTRUM CALCULATION, THE USER MAY ENTER THE LEGENDRE COMPONENTS OF THE ANGULAR FLUX DIRECTLY. FOR AS MANY INPUT-DESIGNATED BROAD GROUP STRUCTURES AS DESIRED, THE CODE CALCULATES AND SAVES (FOR THE COMBINING SECTION) SPECTRUM-WEIGHTED AVERAGES OF MICROSCOPIC AND MACROSCOPIC CROSS SECTIONS AND TRANSFER ARRAYS. SLOWING DOWN SOURCES ARE CALCULATED AND SAVED FOR USE IN THE LOWER ENERGY RANGE. GIVEN BASIC NUCLEAR DATA, THE THERMAL SECTION OF GGC4 DETERMINES A THERMAL SPECTRUM BY EITHER READING IT AS INPUT, BY CALCULATING A MAXWELLIAN SPECTRUM FOR A GIVEN TEMPERATURE, OR BY AN ITERATIVE SOLUTION OF THE P3, B0, P1, OR B1 EQUATIONS FOR AN INPUT BUCKLING. TIME MOMENTS OF THE TIME AND ENERGY-DEPENDENT DIFFUSION EQUATIONS ARE CALCULATED (AS AN OPTION) USING THE INPUT BUCKLING TO REPRESENT LEAKAGE. BROAD GROUP CROSS SECTIONS ARE PREPARED BY AVERAGING FINE GROUP CROSS SECTIONS OVER THE CALCULATED SPECTRA. BROAD GROUP STRUCTURES ARE READ AS INPUT. THE COMBINING SECTION OF GGC4 TAKES THE BROAD GROUP-AVERAGED CROSS SECTIONS FROM THE FAST AND THERMAL PORTIONS OF GGC4 AND FORMS MULTIGROUP CROSS SECTION TABLES. THESE TABLES ARE PREPARED IN STANDARD FORMATS FOR TRANSPORT OR DIFFUSION THEORY CALCULATIONS. IN ADDITION, IT IS POSSIBLE TO USE THE COMBINING SECTION TO PRODUCE MIXTURES NOT USED IN THE SPECTRUM CALCULATION OR TO COMBINE THE RESULTS OF DIFFERENT FAST AND THERMAL SECTION CALCULATIONS AND SO ON. THESE OPTIONS ARE DESCRIBED IN REFERENCE 2.
4. METHOD OF SOLUTION - IN THE FAST SECTION EITHER THE P1 OR THE B1, B2, OR B3 APPROXIMATION IS MADE TO THE TRANSPORT EQUATION USING THE POSITIVE, ENERGY-INDEPENDENT BUCKLING. IN EACH APPROXIMATION LEGENDRE MOMENTS OF THE ANGULAR FLUX ARE COMPUTED BY DIRECT NUMERICAL INTEGRATION OF THE SLOWING DOWN EQUATIONS. IN THE RESC-

4. METHOD OF SOLUTION (CONTINUED)
 NANCE CALCULATIONS, DOPPLER BROADENED (AT AN INPUT TEMPERATURE) ABSORPTION AND SCATTERING CROSS SECTIONS ARE USED. THE RESONANCE TREATMENT ALLOWS UP TO TWO ADMIXED MODERATORS IN AN ABSORBER LUMP IMBEDDED IN A SURROUNDING MODERATOR. THE ABSORBER IN THE LUMP IS TREATED BY USING EITHER THE NARROW RESONANCE APPROXIMATION, THE NARROW RESONANCE INFINITE MASS APPROXIMATION, OR A SOLUTION OF THE SLOWING DOWN INTEGRAL EQUATIONS TO DETERMINE THE COLLISION DENSITY THROUGH THE RESONANCE. THE ADMIXED MODERATORS ARE TREATED BY USING EITHER AN ASYMPTOTIC FORM OF, OR AN INTEGRAL EQUATION SOLUTION FOR, THE COLLISION DENSITY. IN THE RESONANCE CALCULATION EITHER STANDARD GEOMETRY COLLISION PROBABILITIES ARE USED OR TABLES OF COLLISION PROBABILITIES ARE ENTERED. DANCUFF CORRECTIONS CAN ALSO BE MADE. IN THE REGION OF UNRESOLVED RESONANCES, RESONANCE ABSORPTION IS CALCULATED BY USING PORTER-THOMAS DISTRIBUTIONS, BUT ONLY S-WAVE NEUTRONS ARE CONSIDERED. IN THE THERMAL SECTION EITHER THE B0, B1, PC, OR P1 APPROXIMATION TO THE TRANSPORT EQUATION IS MADE, AND IN ALL OPTIONS LEGENDRE MOMENTS OF THE ANGULAR FLUX ARE COMPUTED. A TRAPEZOIDAL ENERGY INTEGRATION MESH IS USED, AND THE RESULTING EQUATIONS ARE SOLVED ITERATIVELY BY USING A SOURCE-NORMALIZED, OVER-RELAXED, GAUSSIAN TECHNIQUE. AVERAGES OVER BROAD GROUPS ARE PERFORMED BY SIMPLE NUMERICAL INTEGRATION. THE RESULTS OBTAINED IN THE FAST AND THERMAL SECTIONS ARE STORED ON SPECIAL TAPES. THESE TAPES MAY CONTAIN RESULTS FOR A NUMBER OF PROBLEMS, EACH PROBLEM INCLUDING FINE GROUP CROSS SECTION DATA FOR A NUMBER OF NUCLIDES. IF THE PROBLEM NUMBER IS SPECIFIED ON THESE TAPES, AND A DESIRED LIST OF NUCLIDES IS GIVEN, THE COMBINING CODE WILL PUNCH MICROSCOPIC CROSS SECTIONS FOR THE REQUESTED LIST OF NUCLIDES. THE PROGRAM ALSO TREATS MIXTURES. GIVEN THE ATOMIC DENSITIES OF THE NUCLIDES IN A MIXTURE, THE CODE WILL PUNCH MACROSCOPIC CROSS SECTIONS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 99 FAST GROUPS
 - 101 THERMAL FINE GROUPS
 - 99 FAST BROAD GROUPS
 - 50 THERMAL BROAD GROUPS
 - 50 BROAD GROUPS IN THE COMBINING SECTION
 - 100 RESONANCES PER NUCLIDE
 - 2 MODERATORS ADMIXED WITH A RESONANCE ABSORBER
 - 305 ENTRIES IN THE ESCAPE PROBABILITY TABLE FOR CYLINDRICAL GEOMETRIES
 - 505 ENTRIES IN THE ESCAPE PROBABILITY TABLE FOR SLAB GEOMETRIES
 A SINGLE AND POSITIVE VALUE FOR THE BUCKLING (B^2) MUST BE SUPPLIED.
6. TYPICAL RUNNING TIME - A B1 CALCULATION IN THE FAST SECTION FOR 3 NUCLIDES AND 6 BROAD GROUPS TAKES APPROXIMATELY 4 MINUTES ON THE UNIVAC1108 IF A RESONANCE CALCULATION (1/2 MINUTE) IS PERFORMED FOR ONE NUCLIDE. THE THERMAL CALCULATION FOR 3 BROAD GROUPS REQUIRES APPROXIMATELY 2 MINUTES, WHICH INCLUDE ABOUT 7 SECONDS FOR THE ITERATIVE PROCEDURE. TO PUNCH STANDARD DIFFUSION AND

6. TYPICAL RUNNING TIME (CONTINUED)
STANDARD TRANSPORT CROSS SECTIONS FOR THIS PROBLEM REQUIRES 2 SECONDS.
7. UNUSUAL FEATURES OF THE PROGRAM - THERE IS AN OPTION IN GGC4 WHICH MAKES IT POSSIBLE TO SHORTEN THE PUNCHING PROCESS FOR LARGE TWO-DIMENSIONAL TRANSFER ARRAYS. THIS CAN BE DONE BY SPECIFYING A MAXIMUM NUMBER OF DESIRED UPSCATTERING AND DOWNSCATTERING TERMS.
8. RELATED AND AUXILIARY PROGRAMS - GGC4 IS A REVISION OF THE EARLIER PROGRAM, GGC3. TO PREPARE, HANDLE, AND UPDATE THE BASIC CROSS SECTION TAPES WHICH ARE USED AS INPUT FOR GGC4, THE FOLLOWING CODES ARE UTILIZED - MAKE, MST, PRINT, MIXER, WTEG, MGT3, SPRINT, COMBIN, AND DOP.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
UNIVAC1108 VERSION OF GGC3 SUBMITTED SEPTEMBER 1967,
DELETED JUNE 1968.
UNIVAC1108 VERSION OF GGC4 SUBMITTED JUNE 1968.
10. REFERENCES - J. ADIR AND K. C. LATHROP, THEORY OF METHODS USED IN GGC-3 MULTIGROUP CROSS SECTION CODE, GA-7156, JULY 1967.
J. ADIR, S. S. CLARK, R. FROELICH, AND L. J. TODT, USERS AND PROGRAMMERS MANUAL FOR GGC-3 MULTIGROUP CROSS SECTION CODE, PARTS 1 AND 2, GA-7157, JULY 1967.
M. K. DRAKE, DESCRIPTION OF AUXILIARY CODES USED IN THE PREPARATION OF DATA FOR THE GGC-3 CODE, GA-7158, AUGUST 1967.
J. ADIR, GGC4 INPUT INSTRUCTIONS, GA MEMORANDUM, JUNE 20, 1968.
BCDCON, GA NOTE.
11. MACHINE REQUIREMENTS - 64K MEMORY WITH 11 TAPE UNITS (SOME OF WHICH MAY BE DRUM AREAS).
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - UNIVAC EXEC II, GAX29.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THERE IS NO RESTRICTION ON THE NUMBER OF PROBLEMS THAT CAN BE RUN CONSECUTIVELY IN EACH SECTION, NOR IS THERE A RESTRICTION ON THE NUMBER OF NUCLIDES PER PROBLEM. WITHOUT USING THE NEW OPTIONS, GGC4 CAN ALSO BE RUN WITH THE GGC3 INPUT INSTRUCTIONS.
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL (4 TAPES)
SOURCE DECKS (GGC4-11,510 CARDS, WTFG-1646 CARDS, MAKE-1033
CARDS, MST-463 CARDS, PRINT-769 CARDS, MIXER-500
CARDS, MGT3-251 CARDS, SPRINT-334 CARDS, COMBIN-
479 CARDS, COP-927 CARDS)
SAMPLE PROBLEM (GGC4-38 CARDS)
FAST DATA TAPES (60 NUCLIDES - TAPE 2 - 2772 BINARY RECORDS,
151 NUCLIDES - TAPE 3 - 7362 BINARY RECORDS)
THERMAL DATA TAPE (222 NUCLIDES - TAPE 4 - 2854 BINARY RECORDS)
REFERENCE REPORTS, MEMORANDUM, AND NOTE
17. CATEGORY - B
KEYWORDS - MULTIGROUP CROSS SECTIONS, AVERAGES, FAST, THERMAL,
SPECTRA, DOPPLER BROADENING, DANCOFF CORRECTION, ANGULAR
DISTRIBUTION, RESONANCE PARAMETERS, MAKE CODES,
MST CODES, PRINT CODES, MIXER CODES, WTFG CODES, MGT3
CODES, SPRINT CODES, COMBIN CODES, DOP CODES

1. NAME OR DESIGNATION OF PROGRAM - LION
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - LION IS A DIGITAL COMPUTER PROGRAM WHICH WILL SOLVE THREE-DIMENSIONAL TRANSIENT AND STEADY-STATE TEMPERATURE DISTRIBUTION PROBLEMS. THE INPUT CONSISTS OF GEOMETRY, PHYSICAL PROPERTIES, BOUNDARY CONDITIONS, INTERNAL HEAT GENERATION RATES, AND COOLANT FLOW RATES AS A FUNCTION OF TIME. IN ADDITION TO SOLVING PROBLEMS OF HEAT CONDUCTION IN A STRUCTURE, LION CAN HANDLE FORCED CONVECTION, FREE CONVECTION, AND RADIATION OR A COMBINATION OF THESE AT THE SURFACE OF THE STRUCTURE. THE OUTPUT CONSISTS OF COMPLETE NODAL TEMPERATURE DISTRIBUTIONS ALONG WITH SURFACE FLUXES AND HEAT TRANSFER COEFFICIENTS. AN OPTION IS INCLUDED IN THE PROGRAM FOR DETERMINING THE MEAN TEMPERATURE IN ANY SPECIFIED SECTION OF THE STRUCTURE.
4. METHOD OF SOLUTION - THE MODELING IS BASED ON THE CONCEPT OF NODAL POINTS CONNECTED BY ONE-DIMENSIONAL THERMAL CONDUCTANCE EQUATIONS IN AS MANY AS SIX DIRECTIONS SIMULTANEOUSLY. THE EXPLICIT, OR FIRST FORWARD DIFFERENCE METHOD, IS THEN USED TO OBTAIN THE SOLUTIONS TO THESE EQUATIONS FOR THE THREE-DIMENSIONAL FIELD. THE TEMPERATURE CHANGE FOR A SURFACE NODE IS CALCULATED BY APPLYING THE OHMS LAW ANALOGY FOR HEAT FLOW TO A NODE WITH NO CAPACITANCE.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
11*D1 + 22*D2 + 20*D3 + 61*D4 + 15*D5 LESS THAN OR EQUAL TO 21000
VARIABLE NUMBER OF SURFACE CONNECTIONS (D1)
VARIABLE NUMBER OF INTERNAL NODES (D2)
VARIABLE NUMBER OF SURFACE NODES (D3)
VARIABLE NUMBER OF BOUNDARY NODES (D4)
VARIABLE NUMBER OF COOLANT NODES (D5)
15 MATERIALS
7*D2 INTERNAL CONNECTIONS
12 COOLANT CHANNELS (EACH WITH 4 SIDES MAXIMUM)
75 PRINTOUTS
7 INTERNAL CONNECTIONS PER INTERNAL NODE
6 SURFACE CONNECTIONS PER SURFACE NODE
30 MEANS AND DIFFERENCES SETS
100 NODES PER SET (WITHOUT EQUIVALENT LINEAR)
50 NODES PER SET (WITH EQUIVALENT LINEAR)
6 TYPES OF MEANS AND DIFFERENCES SETS
5 GRAPHS
6 CURVES PER GRAPH
13 TYPES OF QUANTITIES GRAPHABLE
4 SUB-COOLED NUCLEATE BOILING REGIONS
6. TYPICAL RUNNING TIME - THE APPROXIMATE RUNNING TIME ON THE CDC6600 IS 1 MINUTE PER 10 NODES FOR A STEADY-STATE PROBLEM AND 1.5 MINUTES PER 10 NODES FOR A TRANSIENT PROBLEM.

7. UNUSUAL FEATURES OF THE PROGRAM - SUBCOOLED NUCLEATE BOILING MAY BE CONSIDERED. HOWEVER, THIS CONSIDERATION IS LIMITED TO WATER.
8. RELATED AND AUXILIARY PROGRAMS - LION SUPERCEDES EARLIER IBM704 AND PHILCO 2000 PROGRAMS CALLED TIGER.
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION SUBMITTED OCTOBER 1967, REPLACED BY MODIFIED VERSION AUGUST 1968.
10. REFERENCES - J. R. SCHMID, G. L. LECHLITER, AND W. W. FISCHER, LION TEMPERATURE DISTRIBUTIONS FOR ARBITRARY SHAPES AND COMPLICATED BOUNDARY CONDITIONS, KAPL-M-6532 (EC-57), JULY 27, 1966.
W. W. FISCHER, LION TALKS A USERS MANUAL FOR THE LION THERMAL-STRUCTURAL EVALUATION CODE, KAPL-M-6533 (EC-58), JULY 31, 1967.
E. J. BINNEY, ADDENDUM TO LION REPORT, KAPL-M-6532, JUNE 28, 1968.
R. J. CULLEN, 6600 CALCOMP PLOTTER ROUTINES, KAPL NOTE, JULY 25, 1966.
11. MACHINE REQUIREMENTS - THE PLOTTER ROUTINE APPLE IS USED TO PRODUCE THE INPUT REQUIRED BY THE ASSOCIATED CALCOMP PLOTTER FOR GRAPHICAL OUTPUT.
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE (CCC6600).
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - BECAUSE OF THE LARGE SIZE OF THE LION PROGRAM IT WAS NECESSARY TO SEGMENT THE PROGRAM.
15. NAME AND ESTABLISHMENT OF AUTHOR -
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KNOLLS ATOMIC POWER LABORATORY
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BOX 1072
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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (3125 CARDS)
SAMPLE PROBLEM (158 CARDS)
REFERENCE REPORTS, ADDENDUM, NOTE
17. CATEGORY - F
KEYWORDS - TEMPERATURE DISTRIBUTION, FLUID FLOW, THERMAL, 3-DIMENSIONAL, HEAT TRANSFER, LIQUIDS, VAPORS

1. NAME OR DESIGNATION OF PROGRAM - SAFE-CREEP
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - SAFE-CREEP PERFORMS A VISCO-ELASTIC ANALYSIS OF PLANE OR AXISYMMETRIC COMPOSITE CONCRETE STRUCTURES WITH AGE- AND TEMPERATURE-DEPENDENT CREEP DATA.
4. METHOD OF SOLUTION - THE PROGRAM USES THE FINITE ELEMENT METHOD WHICH REDUCES THE PROBLEM TO THE SOLUTION OF A SYSTEM OF COUPLED INTEGRAL EQUATIONS. THE METHOD OF SOLUTION IS ITERATIVE ON THE SPACE VARIABLE AND UTILIZES THE TRAPEZOIDAL RULE INTEGRATION FOR THE TIME VARIABLE.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
350 NODES
750 ELEMENTS
60 TIME INCREMENTS
6. TYPICAL RUNNING TIME - 90 MINUTES ARE REQUIRED FOR A FULL-CAPACITY PROBLEM.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - GRDWRK (ACC ABSTRACT 296), SAFE-PCRS (ACC ABSTRACT 250), SAFE-PLANE (ACC ABSTRACT 252), SAFE-SHELL (ACC ABSTRACT 253)
9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
UNIVAC1108 VERSION SUBMITTED OCTOBER 1967.
10. REFERENCES - Y. RASHID AND W. ROCKENHAUSER, PRESSURE VESSEL ANALYSIS BY FINITE ELEMENT TECHNIQUE, GA-7810, 1967.
R. D. BROWN, PROPERTIES OF CONCRETE IN REACTOR VESSELS, PAPER NO. 13, CONFERENCE ON PRESTRESSED CONCRETE PRESSURE VESSELS, LONDON, MARCH 13-17, 1967.
F. J. DE ARRIAGA AND Y. R. RASHID, SAFE-CREEP, A COMPUTER PROGRAM FOR THE VISCOELASTIC ANALYSIS OF AXISYMMETRIC AND PLANE CONCRETE STRUCTURES, A USERS MANUAL, GA-8111, JULY 31, 1967.
11. MACHINE REQUIREMENTS - 65K MEMORY AND 13 TAPES
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - CALL TICKER REFERS TO AN 1108 SYSTEM ROUTINE FOR LOGGING JOB TIME USED. A SUBSTITUTE ROUTINE WILL BE REQUIRED FOR USE OF THIS PROGRAM ON ANOTHER MACHINE.

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15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE -

SOURCE DECK (1549 CARDS)
SAMPLE PROBLEM (53 CARDS)
REFERENCE REPORT, GA-8111

17. CATEGORY - I

KEYWORDS - 2-DIMENSIONAL, STRUCTURAL ANALYSIS, ELASTICITY,
STRESSES, SOLIDS

1. NAME OR DESIGNATION OF PROGRAM - FREVAP6
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE FREVAP TYPE OF CODE FOR ESTIMATING THE RELEASE OF LONGER-LIVED METALLIC FISSION PRODUCTS FROM HTGR FUEL ELEMENTS HAS BEEN DEVELOPED TO TAKE INTO ACCOUNT THE COMBINED EFFECTS OF THE RETENTION OF METALLIC FISSION PRODUCTS BY FUEL PARTICLES AND THE RATHER STRONG ABSORPTION OF THESE FISSION PRODUCTS BY THE GRAPHITE OF THE FUEL ELEMENTS. RELEASE CALCULATIONS ARE MADE ON THE BASIS THAT THE LOSS OF FISSION PRODUCT NUCLIDES SUCH AS OF STRONTIUM, CESIUM, AND BARIUM IS DETERMINED BY THEIR EVAPORATION FROM THE GRAPHITE SURFACES AND THEIR TRANSPIRATION INDUCED BY THE FLOWING HELIUM COOLANT. THE CODE IS DESIGNED SO THAT CHANGES OF FISSION RATE (FUEL ELEMENT POWER), FUEL TEMPERATURE, AND GRAPHITE TEMPERATURE MAY BE INCORPORATED INTO THE CALCULATION. TEMPERATURE IS QUITE IMPORTANT IN DETERMINING RELEASE BECAUSE, IN GENERAL, BOTH RELEASE FROM FUEL PARTICLES AND LOSS BY EVAPORATION (TRANSPIRATION) VARY EXPONENTIALLY WITH THE RECIPROCAL OF THE ABSOLUTE TEMPERATURE.
4. METHOD OF SOLUTION - A PROBLEM IS DEFINED AS HAVING A ONE-DIMENSIONAL SEGMENT MADE UP OF THREE PARTS - (1) THE FISSION PRODUCT SOURCE (FUEL PARTICLES) IN SERIES WITH, (2) A NON-SOURCE AND ABSORPTION PART (ELEMENT GRAPHITE) AND (3) A SURFACE FOR EVAPORATION TO THE COOLANT (GRAPHITE-HELIUM INTERFACE). MORE THAN ONE SEGMENT MAY BE CONNECTED (POSSIBLY SEGMENTS STACKED AXIALLY) BY WAY OF THE COOLANT. AT ANY GIVEN SEGMENT, A CONTINUITY EQUATION IS SOLVED ASSUMING EQUILIBRIUM BETWEEN THE SOURCE TERM, ABSORPTION TERM, EVAPORATION AT COOLANT INTERFACE AND THE PARTIAL PRESSURE OF THE FISSION PRODUCT ISOTOPE IN THE COOLANT.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 5 ISOTOPEs
 - 10 TIME INTERVALS FOR TIME-DEPENDENT VARIABLE
 - 49 SEGMENTS*(NUMBER OF ISOTOPEs)
 - 5 DIFFERENT OUTPUT PRINT TIME-STEPS
6. TYPICAL RUNNING TIME - 15 MINUTES ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED OCTOBER 1967.
10. REFERENCE - L. R. ZUMWALT, V. H. PIERCE, AND M. J. GERBER, FREVAP-6 CODE FOR ESTIMATING METALLIC FISSION PRODUCT RELEASE, GAMM-7656, JANUARY 12, 1967.

11. MACHINE REQUIREMENTS - 64K MEMORY WITH 2 TAPES (I/O)
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II, GAX27.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE RKGA SUBROUTINE PROVIDES THE SOLUTION OF THE DERIVATIVE EQUATION BY THE RUNGE-KUTTA-GILL METHOD TO USE AS THE STARTING HAMDE VALUE. THE HAMDE SUBROUTINE IS THEN USED FOR THE SOLUTION OF THE DERIVATIVE EQUATION USING THE MILNE-HAMMING METHOD.
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE -
SOURCE DECK (1104 CARDS)
SAMPLE PROBLEM (95 CARDS)
REFERENCE REPORT
17. CATEGORY - C
KEYWORDS - FISSION PRODUCTS, HTGR REACTORS, FUEL ELEMENTS,
1-DIMENSIONAL, ISCTOPES, TEMPERATURE

1. NAME OR DESIGNATION OF PROGRAM - GAFFE
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - A ZERO-DIMENSIONAL CALCULATION OF FEED FUEL REQUIREMENTS IS PERFORMED TO PRODUCE A SPECIFIED END OF CYCLE MULTIPLICATION FACTOR FOR THE EQUILIBRIUM FUEL CYCLE, GIVEN FEED COMPOSITION, LENGTH OF CYCLE AND REACTOR POWER. IT IS ALTERNATELY POSSIBLE TO COMPUTE CYCLE LENGTH OR FEED ENRICHMENT. THE CODE IS A SURVEY TOOL WHICH ASSUMES PERIODIC REFUELING AND PERMITS COMPLETE OR PARTIAL RECYCLING OF MATERIALS. THE SEGREGATED FUEL CONCEPT CAN BE HANDLED WITHIN THE FRAMEWORK OF THE CALCULATION.
4. METHOD OF SOLUTION - DEPLETION IS COMPUTED FOR A SINGLE BATCH OF FUEL OVER THE ENTIRE PERIOD IT REMAINS IN THE REACTOR USING FLUXES WHICH ARE CHARACTERISTIC OF THE AVERAGE COMPOSITION OF THE EQUILIBRIUM REACTOR. BEGINNING WITH AN INITIAL GUESS FOR FEED OR CYCLE TIME, A DOUBLE SET OF ITERATIONS ARE PERFORMED. THE OUTER ITERATIONS COMPRISE THE FEED OR TIME SEARCH TO OBTAIN THE DESIRED END OF CYCLE KEFF. THE INNER ITERATIONS CONVERGE THE AVERAGE CYCLE FLUX AND RECYCLE COMPOSITION, GIVEN A SPECIFIC FEED AND CYCLE LENGTH. ONCE THE SEARCHES ARE SATISFIED, THEN VARIOUS NEUTRONIC CHARACTERISTICS OF THE EQUILIBRIUM CYCLE CAN BE CALCULATED DIRECTLY.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROGRAM - MAXIMA OF -
18 ENERGY GROUPS
50 TOTAL NUCLIDES
3 NUCLIDES WITH FULL TRANSFER MATRIX
6. TYPICAL RUNNING TIME - 30 SECONDS ARE REQUIRED FOR A 15 ENERGY GROUP AND 39 NUCLIDE PROBLEM.
7. UNUSUAL FEATURES OF THE PROGRAM - GAFFE COMPUTES THE CHARACTERISTICS OF THE EQUILIBRIUM CYCLE WITHOUT GOING THROUGH AN APPROACH TO EQUILIBRIUM CALCULATION WHICH WOULD REQUIRE FOLLOWING (DEPLETING) A NUMBER OF INDIVIDUAL SETS OF FUEL COMPOSITIONS EQUAL TO AT LEAST ONE SET FOR EACH REFUELING WHICH SHOULD OCCUR DURING THE ENTIRE RESIDENCE PERIOD OF A SINGLE BATCH.
8. RELATED AND AUXILIARY PROGRAMS - GAFFE PUNCHES CARDS FOR THE WAMPUM EQUILIBRIUM FUEL CYCLE COST CODE (ACC ABSTRACT 224). THE VALANC CODE HAS BEEN ATTACHED AS A SUBROUTINE.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED OCTOBER 1967.
10. REFERENCE -
11. MACHINE REQUIREMENTS - 65K MEMORY AND 2 TAPES (I/O)

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12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
EXEC II, GAX27.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (3871 CARDS)
17. CATEGORY - C
KEYWORDS - DEPLETION, FUEL CYCLE, MULTIGROUP, REACTIVITY, SEARCHES

1. NAME OR DESIGNATION OF PROGRAM - BLOOST6
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - BLOOST6 COMBINES A REACTOR SPACE-INDEPENDENT KINETICS CODE WITH A TWO-DIMENSIONAL HEAT TRANSFER CODE, AND A TIME-DEPENDENT SPHERICAL GEOMETRY HEAT TRANSFER ROUTINE FOR FUEL PARTICLES. THE CODE IS APPLICABLE TO PROBLEMS FOR WHICH THE SPACE-INDEPENDENT FORM OF THE REACTOR KINETICS EQUATION IS APPLICABLE.
4. METHOD OF SOLUTION - A FOURTH-ORDER RUNGE-KUTTA INTEGRATION IS USED FOR THE KINETICS EQUATION. A PEACEMAN-RACHFORD ITERATION SCHEME IS EMPLOYED TO SOLVE THE TWO-DIMENSIONAL TIME-DEPENDENT HEAT CONDUCTION EQUATION IN R-Z GEOMETRY. COUPLING THE R-Z FUEL ELEMENT MODEL TO THE SPHERICAL GEOMETRY HEAT TRANSFER ROUTINE FOR THE FUEL PARTICLES IS ACCOMPLISHED THROUGH A HEAT TRANSFER COEFFICIENT, SPECIFIED BY AN EASILY MODIFIED FORTRAN FUNCTION STATEMENT.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - TWO-DIMENSIONAL HEAT TRANSFER ROUTINES ARE LIMITED TO R-Z GEOMETRY, 36 REGIONS, 19 MATERIALS, 15 RADIAL MESH POINTS, 20 AXIAL MESH POINTS, 6 RADIAL REGION BOUNDARIES, 10 AXIAL REGION BOUNDARIES, AND INNER AND OUTER AXIAL CCLANTS. KINETICS EQUATIONS ALLOW 6 DELAYED GROUPS AND 2 TEMPERATURE COEFFICIENTS. 2 TYPES OF FUEL PARTICLES ARE PERMITTED.
6. TYPICAL RUNNING TIME - FOR A STEP CHANGE IN REACTIVITY OF SEVERAL DOLLARS AND A FAIRLY DETAILED THERMODYNAMIC MODEL OF THE FUEL ELEMENT, 5 TO 10 MINUTES ARE REQUIRED TO FOLLOW THE EXCURSION FOR 10 TO 15 SECONDS OF REAL TIME.
7. UNUSUAL FEATURES OF THE PROGRAM - THE MATERIAL, GAS, AND COOLANT PROPERTIES AND HEAT GENERATION RATES ARE SUPPLIED AS FORTRAN ARITHMETIC STATEMENT FUNCTIONS IN A SUBROUTINE CALLED DATA. THIS SUBROUTINE IS THEREFORE UNIQUE TO A PARTICULAR FUEL ELEMENT DESIGN AND MAY BE CONSIDERED AS PART OF THE INPUT DATA. FOR CASES IN WHICH THE INITIAL TEMPERATURE DISTRIBUTION IS UNKNOWN IT IS POSSIBLE TO RUN THE BLOOST CODE AS A PURE HEAT TRANSFER CODE OMITTING REACTIVITY INPUT TO THE KINETICS PORTION AND TO OBTAIN A TEMPERATURE DISTRIBUTION APPROPRIATE TO INITIAL POWER CONDITIONS PUNCHED ON CARDS SUITABLE FOR LATER USE IN KINETICS PROBLEMS.
8. RELATED AND AUXILIARY PROGRAMS - BLOOST5 (ACC ABSTRACT 205), AN EARLIER LINKAGE OF THE KINETICS PORTION AND THE TWO-DIMENSIONAL HEAT TRANSFER ROUTINE. RAT, AN EARLIER STAND-ALONE HEAT TRANSFER CODE.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED OCTOBER 1967.

10. REFERENCES - R. C. DAHLBERG AND M. H. MERRILL, BLOOST-6 - A KINETICS CODE CONTAINING A THERMODYNAMIC MODEL OF COATED PARTICLES FOR HTGR APPLICATIONS, GAMO-8119, JULY 19, 1967.
M. H. MERRILL, BLOOST-6, A COMBINED REACTOR KINETICS HEAT TRANSFER PROGRAM, GA-8416, DECEMBER 15, 1967.
11. MACHINE REQUIREMENTS - 64K MEMORY WITH 2 TAPES (I/O)
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II, GAX27.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
M. H. MERRILL
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (6693 CARDS)
SAMPLE PROBLEM (484 CARDS)
REFERENCE REPORTS
17. CATEGORY - E
KEYWORDS - SPACE-INDEPENDENT KINETICS, TEMPERATURE DISTRIBUTION, THERMODYNAMICS, 2-DIMENSIONAL, R-Z GEOMETRIES, HEAT TRANSFER, FUEL ELEMENTS, RAT CODES, FUGRA CODES, HTGR REACTORS

1. NAME OR DESIGNATION OF PROGRAM - PERT4
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - PERT4 COMPUTES REACTIVITY COEFFICIENT TRAVERSES IN X-Y, R-Z, OR R-THETA GEOMETRY USING THE FIRST-ORDER PERTURBATION EQUATIONS IN THE DIFFUSION APPROXIMATION. FLUX AND ADJOINT INPUT CAN BE TAKEN DIRECTLY FROM 2-D CALCULATIONS OR SYNTHESIZED FROM RADIAL AND AXIAL 1-D CALCULATIONS. THE CODE CAN ALSO BE USED TO COMPUTE ACTIVITY TRAVERSES FOR ANY CROSS SECTION OF ANY MATERIAL, THE NEUTRON GENERATION TIME, AND THE EFFECTIVE DELAYED NEUTRON FRACTION.
4. METHOD OF SOLUTION - STANDARD DIFFERENCE TECHNIQUES ARE USED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - PERT4 WILL HANDLE UP TO 26 ENERGY GROUPS WITH A 50 X 50 SPATIAL MESH.
6. TYPICAL RUNNING TIME - ABOUT 30 SECONDS PER CASE ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - THE INPUT DATA (GEOMETRY, DENSITIES, CROSS SECTIONS) IS COMPATIBLE WITH THE LOS ALAMOS TRANSPORT CODES DTF4 (ACC ABSTRACT 209) AND 2DF (ACC ABSTRACT 173).
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED OCTOBER 1967.
10. REFERENCE - R. W. HARDIE AND W. W. LITTLE, JR., PERT-IV, A TWO-DIMENSIONAL PERTURBATION CODE IN FORTRAN-IV, BNWL-409, APRIL 1967.
11. MACHINE REQUIREMENTS - 65K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
W. W. LITTLE, JR.
BATTELLE-NORTHWEST LABORATORY
P. O. BOX 999
RICHLAND, WASHINGTON 99352
16. MATERIAL AVAILABLE -
SOURCE DECK (985 CARDS)
SAMPLE PROBLEM (366 CARDS)
REFERENCE REPORT

17. CATEGORY - C
KEYWORDS - PERTURBATION THEORY, FAST REACTORS, 2-DIMENSIONAL,
MULTIGROUP, REACTIVITY, X-Y, R-Z, R-THETA GEOMETRIES,
DTF4 CODES, 2DF CODES

1. NAME OR DESIGNATION OF PROGRAM - STRIP
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - PHILCO 2000
3. NATURE OF PHYSICAL PROBLEM SOLVED - CAPTURE AND FISSION RESONANCE INTEGRALS ARE CALCULATED BY A FAST METHOD IN THE RESOLVED RESONANCE RANGE, TAKING EXPLICIT ACCOUNT OF OVERLAP AND INTERFERENCE BETWEEN RESONANCES IN A MIXTURE OF RESONANCE ABSORBERS. THE RESONANCE INTEGRALS ARE CALCULATED OVER ARBITRARY ENERGY BANDS. OPTIONALLY, THE NEUTRON FLUX AS A FUNCTION OF ENERGY IN ONE OR TWO REGIONS MAY BE PRINTED OUT.
4. METHOD OF SOLUTION - THE NEUTRON FLUX IS CALCULATED IN TWO REGIONS AS A FUNCTION OF ENERGY BY SOLVING TWO COUPLED INTEGRAL SLOWING DOWN EQUATIONS, USING FIRST FLIGHT COLLISION PROBABILITIES BASED ON A FLAT FLUX APPROXIMATION. IN ORDER TO MEET DESIGN REQUIREMENTS FOR A RAPID AND ACCURATE CROSS SECTION PROGRAM WITH A COMPLETE TREATMENT OF A LARGE NUMBER OF RESONANCES, SEVERAL COMPUTATIONAL AIDS HAVE BEEN INCORPORATED, THE TWO MOST IMPORTANT IMPROVEMENTS BEING THE CALCULATION OF THE SLOWING DOWN SOURCE BY A RECURRENCE RELATION, AND THE USE OF NON-CONSTANT FINE ENERGY GROUP WIDTHS, WHICH ARE AUTOMATICALLY CALCULATED TAKING INTO ACCOUNT THE RATE OF CHANGE OF THE CROSS SECTIONS AND THE RELATIVELY HIGHER IMPORTANCE OF THE LOW-LYING RESONANCES WITH RESPECT TO THEIR CONTRIBUTION TO THE TOTAL RESONANCE INTEGRAL. THE DOPPLER-BROADENED RESONANCE CROSS SECTIONS ARE CALCULATED USING A SINGLE-LEVEL BREIT-WIGNER FORMULA. THE CAPTURE AND FISSION CROSS SECTIONS MAY BE CORRECTED BY A SET OF TABULATED SMOOTH CORRECTION TERMS. THE SCATTERING IS ASSUMED TO BE ISOTROPIC IN THE CENTER-OF-MASS SYSTEM.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE NUMBER OF REGIONS ALLOWED IS ONE OR TWO. IN A TWO-REGION PROBLEM, THE INNER REGION MAY HAVE THE SHAPE OF AN INFINITE CYLINDER, AN INFINITE SLAB, OR A SPHERE. THE FLUX IN THE OUTER REGION MAY BE COMPUTED OR FORCED TO BE $1/E$. THE MAXIMUM NUMBER OF ELEMENTS, EACH OF WHICH MAY APPEAR IN ONE OR TWO REGIONS, IS TEN. THREE OF THESE ELEMENTS MAY INCLUDE RESONANCES. EACH ELEMENT MAY CONTAIN ANY NUMBER OF ISOTOPES WITHIN THE LIMIT OF A TOTAL NUMBER OF 15 ISOTOPES. THE MAXIMUM TOTAL NUMBER OF RESONANCES IS 500. THE PERMISSIBLE NUMBER OF ENERGY POINTS FOR SMOOTH CROSS SECTION CORRECTION IS 100, AND 100 BROAD ENERGY GROUPS ARE ALLOWED.
6. TYPICAL RUNNING TIME - TO CALCULATE THE RESONANCE INTEGRAL OF ${}^{235}\text{U}$ BELOW 4 KEV, INCLUDING 228 RESOLVED RESONANCES, TO AN ACCURACY OF 0.1 PER CENT, THE RUNNING TIME IS ABOUT 3 MINUTES. FOR THIS PARTICULAR CALCULATION, 2400 FINE ENERGY GROUPS WERE UTILIZED.
7. UNUSUAL FEATURES OF THE PROGRAM -

REACTOR CODE ABSTRACT 305

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8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
PHILCO 2000 VERSION SUBMITTED OCTOBER 1967.
10. REFERENCES - J. HELHOLTZ AND D. H. ROY, STRIP - RESONANCE ABSORPTION PROGRAM TREATING OVERLAP AND INTERFERENCE, TP-332, SEPTEMBER 1967.
J. HELHOLTZ AND D. H. ROY, RESONANCE INTEGRAL CALCULATIONS TREATING OVERLAP AND INTERFERENCE, TRANSACTIONS AMERICAN NUCLEAR SOCIETY, 10, NO. 2, NOVEMBER 1967.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
J. HELHOLTZ AND D. H. ROY
BABCOCK AND WILCOX
P. O. BOX 1260
LYNCHBURG, VIRGINIA 24505
16. MATERIAL AVAILABLE -
SOURCE DECK (1139 CARDS)
REFERENCE REPORT, TP-332
17. CATEGORY - A
KEYWORDS - RESONANCE INTEGRALS, RESOLVED REGION, CAPTURE, FISSION, CROSS SECTIONS, BREIT-WIGNER FORMULA, DOPPLER BROADENING, DANCOFF CORRECTION, RESONANCE ESCAPE PROBABILITY

1. NAME OR DESIGNATION OF PROGRAM - FCC4
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - FCC4 IS A MULTIPURPOSE DATA MANIPULATION CODE FOR USE IN FAST REACTOR ANALYSIS. THE CODE CAN BE USED TO - (A) COMPUTE RESONANCE-SHIELDED CROSS SECTIONS USING DATA IN THE RUSSIAN FORMAT (SHIELDING FACTORS AND INFINITE-DILUTION CROSS SECTIONS), (B) COMPUTE MULTIGROUP FUNDAMENTAL-MODE FLUX AND ADJACENT FLUX, (C) COMPUTE AND PUNCH GROUP-COLLAPSED MICROSCOPIC OR MACROSCOPIC CROSS SECTIONS IN THE DTF FORMAT, (D) COMPUTE FUEL BURNUP AT CONSTANT FLUX OR POWER DENSITY.
4. METHOD OF SOLUTION -
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - FCC4 WILL HANDLE UP TO 40 ENERGY GROUPS AND 10 DOWNSCATTERING TERMS WITH 14 ISOTOPES.
6. TYPICAL RUNNING TIME - ABOUT 15 SECONDS PER CASE ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM - AN EXTENSIVE CROSS CHECK (FOR CONSISTENCY) OF THE INPUT DATA IS PERFORMED. BY THIS CHECK, NUMEROUS PRINTING ERRORS IN THE RUSSIAN DATA COMPILATION HAVE BEEN DETECTED.
8. RELATED AND AUXILIARY PROGRAMS - FCC4 IS AN EXTENSIVE REVISION OF THE FCC CODE. THE CROSS SECTION TAPE IS PREPARED BY THE PUP-IV CODE.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED OCTOBER 1967.
10. REFERENCES - W. W. LITTLE, JR. AND R. W. HARDIE, FCC-IV, A REVISED VERSION OF THE FCC FUNDAMENTAL MODE FAST REACTOR CODE, BNWL-450, AUGUST 1967.
I. I. BONDARENKO ET AL, GROUP CONSTANTS FOR NUCLEAR REACTOR CALCULATIONS, CONSULTANTS BUREAU, NEW YORK, 1964.
11. MACHINE REQUIREMENTS - 65K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

REACTOR CODE ABSTRACT 306

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15. NAME AND ESTABLISHMENT OF AUTHORS -

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BATTELLE-NORTHWEST LABORATORY
P. O. BOX 999
RICHLAND, WASHINGTON 99352

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECK (2204 CARDS)
SAMPLE PROBLEM (196 CARDS)
REFERENCE REPORT, BNWL-450

17. CATEGORY - B

KEYWORDS - FAST REACTORS, FLUX SPECTRUM, AVERAGES, CROSS SECTIONS,
MULTIGROUP, DEPLETION, PUP4 CODES

1. NAME OR DESIGNATION OF PROGRAM - HWCCR-SAFE
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360
3. NATURE OF PHYSICAL PROBLEM SOLVED - HWCCR-SAFE IS A MONTE CARLO THERMAL REACTOR ANALYSIS PROGRAM DESIGNED FOR USE WITH PROPOSED HWCCR LATTICE CONFIGURATIONS. IT IS A BENCHMARK TOOL TO CHECK MULTIGROUP DIFFUSION AND TRANSPORT CALCULATIONS AND TO EVALUATE THE EFFECT OF THEIR USE OF GEOMETRIC APPROXIMATIONS.
4. METHOD OF SOLUTION - NEUTRONS ARE STARTED FROM AN INITIAL SOURCE DISTRIBUTION IN SPACE, ANGLE, AND ENERGY WITH A STATISTICAL WEIGHT OF UNITY. THEY ARE FOLLOWED FROM COLLISION TO COLLISION UNTIL THEY ARE EITHER CAPTURED OR PRODUCE FISSION. K-INFINITY IS THE AVERAGE NUMBER OF NEUTRONS PRODUCED PER SOURCE NEUTRON. THE SELECTION OF THE INITIAL ENERGY IS BASED ON ASSUMING A SPECTRUM FOR FISSION NEUTRONS AND TABULATING THE AVERAGE ENERGY BETWEEN ENERGY POINTS DEFINING EQUAL PROBABILITY INTERVALS. THE CRANBERG SPECTRUM FOR U235 HAS BEEN USED. THE DIRECTION COSINES DEFINING PARTICLE MOTION ARE SELECTED ISOTROPICALLY FOR THE EMERGENCE OF A NEUTRON FROM A FISSION REACTION. THE MEAN FREE PATH OF THE NEUTRON AT ITS PRESENT ENERGY DETERMINES THE NEXT COLLISION. THE DETERMINATION OF THE NEUTRON ENERGY SUBSEQUENT TO A COLLISION IS BASED ON THE TYPE OF INTERACTION WHICH OCCURRED. HWCCR-SAFE DIVIDES THE 0-10 MEV ENERGY RANGE INTO FAST (1 KEV TO 10 MEV), RESONANCE (1EV TO 1 KEV), AND THERMAL (0 TO 1 EV) RANGES. ELASTIC COLLISIONS IN THE FAST AND RESONANCE RANGES ASSUME ISOTROPIC SCATTERING. THE EVAPORATION MODEL IS USED FOR INELASTIC COLLISIONS IN THE FAST RANGE. A MODIFIED FREE ATOM MODEL IS USED FOR ELASTIC SCATTERING IN THE THERMAL ENERGY RANGE. COLLISION-TYPE ESTIMATORS ARE USED EXCEPT FOR THIN REGIONS WHERE STATISTICAL ESTIMATION IS SUPPLEMENTED WITH AN IMPROVED TREATMENT OF SCATTERING COLLISIONS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE HWCCR-SAFE PROGRAM IS A TWO-DIMENSIONAL CELL CODE IN WHICH THE CELL IS ASSUMED TO BE RECTANGULAR. UP TO 20 MATERIAL REGIONS (ZONES) CAN BE SPECIFIED WHERE ONE OF THE ZONES IS COMPLETELY DESCRIBED BY THE CELL BOUNDARIES. ALL OTHER ZONES ARE ASSIGNED CONCENTRIC CYLINDERS WHICH LIE WITHIN A CYLINDER DESCRIBED AS OUTER WALL BOUNDARY, OR WITHIN THE RIGHT CIRCULAR CYLINDRICAL RODS WHICH ARE INSERTED IN THE INNERMOST CYLINDER DESIGNATED AS THE INNER WALL BOUNDARY. UP TO 40 RODS MAY BE INSERTED, AND 6 TYPES OF ROD MAY BE SPECIFIED.
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - THE HWCCR-SAFE PROGRAM IS AN ADAPTATION AND EXTENSION OF THE ATOMICS INTERNATIONAL SAFE CODE

WRITTEN TO HANDLE IDENTICAL CYLINDRICAL RODS IN THREE-DIMENSIONAL FINITE OR INFINITE ARRAYS.

9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM360 VERSION SUBMITTED OCTOBER 1967.
10. REFERENCES - L. B. LEVITT AND R. C. LEWIS, HWOCR-SAFE, A MONTE CARLO REACTOR ANALYSIS CODE, AI-CE-MEMO-69, JUNE 30, 1967.
RELICS INPUT DESCRIPTION, AI NOTE.
INPUT DESCRIPTION FOR HWOCR-SAFE CROSS SECTION TAPE MAKER, AI NOTE.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN IV (H)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - CS/360.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - VARIATIONS OF THIS PROGRAM HAVE BEEN PREPARED FOR THE CDC6600 AND IBM7094 COMPUTERS.
15. NAME AND ESTABLISHMENT OF AUTHORS -
L. B. LEVITT AND R. C. LEWIS
HWOCR PROGRAM OFFICE
P. O. BOX 309
CANOGA PARK, CALIFORNIA 91304
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (SAFE 2682 CARDS, RELICS 76 CARDS, TAPE-MAKER 262 CARDS)
SAMPLE PROBLEMS (SAFE 245 CARDS, RELICS 30 CARDS, TAPE-MAKER 1835 CARDS)
REFERENCE REPORT AND NOTES
17. CATEGORY - B
KEYWORDS - CELL CALCULATION, 2-DIMENSIONAL, MONTE CARLO METHOD, HWOCR REACTORS, REACTIVITY, RELICS CODES, TAPE-MAKER CODES

1. NAME OR DESIGNATION OF PROGRAM - JUPITOR1(JP1)
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC1604
3. NATURE OF PHYSICAL PROBLEM SOLVED - THIS IS A PROGRAM FOR PERFORMING COUPLED-CHANNEL CALCULATIONS TO EVALUATE THE CROSS SECTIONS FOR THE SCATTERING OF NUCLEAR PARTICLES BY VARIOUS COLLECTIVE NUCLEI.
4. METHOD OF SOLUTION - IN SOLVING THE COUPLED DIFFERENTIAL EQUATIONS THE PREDICTION-CORRECTION METHOD DUE TO STORMER IS USED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - A MAXIMA OF SIX STATES AND 30 PARTIAL WAVES CAN BE COUPLED AT ONE TIME. IF THE PROJECTILE ENERGY BECOMES NEGATIVE IN SOME EXCITED CHANNELS THIS MAXIMUM IS REDUCED TO 25 PARTIAL WAVES. THE MAXIMUM VALUE OF THE ORBITAL ANGULAR MOMENTUM IS 69. DIFFERENTIAL CROSS SECTIONS FOR UP TO 100 ANGLES FOR ANY NUMBER OF STATES CAN BE COMPUTED UNLESS A POLARIZED BEAM OR TARGET IS CONSIDERED, IN WHICH CASE THE MAXIMUM IS 35.
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM - SPIN OF THE PROJECTILE CAN BE 0, 1/2, OR 1 FOR NON-ADIABATIC CALCULATIONS, 0 OR 1/2 FOR ADIABATIC. TARGETS CAN BE EITHER VIBRATIONAL (SPHERICAL) OR ROTATIONAL (PERMANENTLY DEFORMED) AND OF EITHER EVEN OR ODD A. WHEN THE TARGET IS DEFORMED, EXCITATION OF STATES BELONGING TO HIGHER (VIBRATIONAL) BANDS CAN BE CONSIDERED. COULOMB EXCITATION CAN BE INCLUDED, AND THE FORM FACTOR CAN BE EITHER REAL OR COMPLEX.
8. RELATED AND AUXILIARY PROGRAMS - JUPITOR2 IS UNDER PREPARATION TO HANDLE SMALLER SETS OF EQUATIONS MORE EFFICIENTLY.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC1604 VERSION SUBMITTED NOVEMBER 1967.
10. REFERENCES - T. TAMURA, COMPUTER PROGRAM JUPITOR-1 FOR COUPLED-CHANNEL CALCULATIONS, ORNL-4152, AUGUST 1967, ADDENDUM I, AUGUST 14, 1967, ADDENDA/CORRIGENDA NO. 2, NOVEMBER 27, 1967.
T. TAMURA, REVS. MOD. PHYS., VOL. 37, P. 679, 1965.
11. MACHINE REQUIREMENTS - CDC1604 WITH 32K CORE MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN 63
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -

TARO TAMURA
PHYSICS DIVISION
OAK RIDGE NATIONAL LABORATORY
P. O. BOX X
OAK RIDGE, TENNESSEE 37831

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECK (2954 CARDS)
SAMPLE PROBLEM (92 CARDS)
REFERENCE REPORT, ORNL-4152 AND ADDENDA

17. CATEGORY - A

KEYWORDS - SCATTERING, CROSS SECTIONS, COMPOUND NUCLEUS, ANGULAR
DISTRIBUTION

1. NAME OR DESIGNATION OF PROGRAM - TSN
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7094,360
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE TIME-DEPENDENT NEUTRON TRANSPORT EQUATION IS SOLVED. ENERGY DEPOSITION IS ALLOWED TO CAUSE VARIATION IN THE NEUTRON CROSS SECTIONS FOR THE CORE REGION, IN THE CORE DENSITY, AND IN CORE HEIGHT. THE RESULT IS A KINETICS CALCULATION INCLUDING SPATIAL DEPENDENCE BOTH IN FEEDBACK EFFECTS AND IN NEUTRON DENSITY. THE RESULTS ARE SUMMARIZED IN A MANNER SIMILAR TO POINT-KINETICS CODES, AND SPATIAL DISTRIBUTIONS ARE ALSO GIVEN. GRAPHICAL SUMMARIES OF THE SIGNIFICANT VARIABLES AND SPATIAL DISTRIBUTIONS ARE GIVEN. IN THE IBM360 VERSION, EITHER THE NEUTRON YIELD FROM FISSION OR THE THICKNESS OF A SPECIFIED ZONE CAN BE CHANGED AS AN INDEPENDENT FUNCTION OF TIME AS SPECIFIED BY THE USER. THIS PROVIDES TWO WAYS OF ALLOWING EXPLICIT REACTIVITY VARIATION WITH TIME.
4. METHOD OF SOLUTION - CARLSON'S SN METHOD IS USED TO SOLVE THE NEUTRON TRANSPORT EQUATION. ANALOGOUS FINITE-DIFFERENCE TECHNIQUES ARE USED TO SOLVE THE FEEDBACK EQUATIONS. THE VARIATIONS IN CROSS SECTIONS AND DENSITIES WITH ENERGY DEPOSITION ARE EVALUATED FROM POLYNOMIAL COEFFICIENTS SUPPLIED BY THE USER.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - 2 VELOCITY GROUPS, 100 SPATIAL INTERVALS, 1-DIMENSIONAL GEOMETRY S4 (CYLINDRICAL) OR S6 (PLANE OR SPHERE). IN THE IBM7094 VERSION, THE CORE MUST BE THE FIRST ZONE AND MUST BE HOMOGENEOUS. THE $1/V$ TERM IS NEGLECTED FOR THE FAST GROUP. THESE LAST RESTRICTIONS HAVE BEEN REMOVED IN THE IBM360 VERSION.
6. TYPICAL RUNNING TIME - A TYPICAL PROBLEM SUPPLIED WITH THE DOCUMENTATION RAN 5.2 MINUTES ON THE IBM7094 AND 18.1 MINUTES ON THE IBM360, MODEL 50. THE LATTER TIME CAN BE REDUCED BY APPLYING THE OPTIMIZATION FEATURE OF FORTRAN IV LEVEL H, WHERE AVAILABLE.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - TSN IS BASED ON AND USES PORTIONS OF ANISN (ACC ABSTRACT 151).
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM7094 VERSION SUBMITTED NOVEMBER 1967.
IBM360 VERSION SUBMITTED NOVEMBER 1967.
10. REFERENCES - W. A. RHOADES AND J. G. MORGAN, THE TSN COMPUTER CODE FOR SPATIALLY DEPENDENT REACTOR KINETICS CALCULATIONS, NAA-SR-MEMO-12319, APRIL 18, 1967.
M. A. BOLING AND W. A. RHOADES, AICRT3-3, SPRNTV, AND FLINGV, GENERAL PURPOSE SUBROUTINES FOR DISPLAY OF DIGITAL DATA, NAA-SR-TDR-11915, APRIL 1966.

10. REFERENCES (CONTINUED)

NOTES ON TSN FOR THE IBM S/360, AI NOTE.

R. A. BLAINE, ATOMICS INTERNATIONAL ENVIRONMENT REPORT, MAY 1967.

W. W. ENGLE, JR., A USERS MANUAL FOR ANISN, A ONE-DIMENSIONAL DISCRETE ORDINATES TRANSPORT CODE WITH ANISOTROPIC SCATTERING, K-1693, MARCH 30, 1967.

M. A. BOLING AND W. A. RHOADES, ANISN/DTF II, CONVERSION TO IBM SYSTEM/360, AI-66-MEMC-171, APRIL 1967.

J. F. JACKSON, W. A. RHOADES, AND L. I. MOSS, ANALYSIS OF SNAPTRAN-1 AND -2 REACTOR KINETICS EXPERIMENTS, NAA-SR-11850, JUNE 26, 1967.

B. G. CARLSON AND K. D. LATHROP, TRANSPORT THEORY - THE METHOD OF DISCRETE ORDINATES, LAMS-3251-MS, OCTOBER 22, 1965.

11. MACHINE REQUIREMENTS - 32K IBM7094 OR IBM360, MODEL 50H OR LARGER

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - FIB III (IBM7094) OR OS/360 (IBM360).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - TSN USES THE AICRT3-3 ROUTINE (REFERENCE 2) AND THE NAA SC-4020 PACKAGE AVAILABLE THROUGH -

LAICE LIBRARIAN

C/O STROMBERG-CARLSON

P. O. BOX 2449

SAN DIEGO, CALIFORNIA

THE USER CAN DELETE THE CRT FEATURE BY SUPPLYING DUMMY SUBROUTINES.

15. NAME AND ESTABLISHMENT OF AUTHOR -

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ENGINEERING ANALYSIS GROUP

ATOMICS INTERNATIONAL

P. O. BOX 309

CANOGA PARK, CALIFORNIA 91304

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (7094-2058 CARDS, 360-1909 CARDS)

SAMPLE PROBLEMS (7094-62 CARDS, 360-62 CARDS)

REFERENCE REPORT, NAA-SR-MEMO-12319, ENVIRONMENT REPORT,
AND AI NOTE

17. CATEGORY - F

KEYWORDS - 2-GROUP, SN METHOD, SPACE-TIME KINETICS, TEMPERATURE COEFFICIENTS, FUELS EXPANSION, FEEDBACK, GRAPHS, 1-DIMENSIONAL, SNAP REACTORS, ANISN CODES

1. NAME OR DESIGNATION OF PROGRAM - GAKIN
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE MULTIGROUP, 1-DIMENSIONAL, TIME-DEPENDENT DIFFUSION THEORY EQUATIONS ARE SOLVED IN SLAB, CYLINDRICAL OR SPHERICAL GEOMETRY WITH DELAYED NEUTRONS TAKEN INTO ACCOUNT. AN ARBITRARY SCATTERING MATRIX IS ALLOWED, TOGETHER WITH A PIECE-WISE, LINEAR, TIME-DEPENDENT, INHOMOGENEOUS SOURCE TERM. FEEDBACK IS AVAILABLE FROM TIME-DEPENDENT CROSS SECTION CHANGES AND BUILDUP IN XENON. THE TIME INTEGRATION IS DIVIDED INTO TIME ZONES WITH UNIQUE FEEDBACK AND SOURCE DATA FOR EACH ZONE.
4. METHOD OF SOLUTION - THE BASIC ASSUMPTION FOR THE INTEGRATION OF THE TIME-DEPENDENT DIFFUSION THEORY EQUATION IS THAT THE NEUTRON AND PRECURSOR DENSITIES BEHAVE EXPONENTIALLY DURING SMALL TIME INTERVALS. THIS ASSUMPTION PERMITS A MATRIX FACTORIZATION WHICH GUARANTEES ASYMPTOTIC UNCONDITIONAL STABILITY. THE TIME-STEP SIZE IS AUTOMATICALLY ADJUSTED TO GIVE APPROXIMATELY THE SAME ACCURACY PER TIME-STEP.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE MATHEMATICAL FORMULATION OF THE GAKIN ALGORITHM ALLOWS AN ARBITRARY NUMBER OF GROUPS AND MESH POINTS. THE COMPLETELY CORE CONTAINED PROGRAM HAS THE FOLLOWING RELATIVE RESTRICTIONS -
 - 100 SPACE POINTS
 - 20 HOMOGENEOUS REGIONS
 - 10 FLUX GROUPS
 - 6 DELAYED NEUTRON GROUPS
6. TYPICAL RUNNING TIME -

SPACE POINTS	FLUX GROUPS	DELAYED GROUPS	SECONDS PER TIME-STEP
21	2	1	.016
49	2	1	.050
75	2	1	.062
7. UNUSUAL FEATURES OF THE PROGRAM -
 - (A) ASYMPTOTIC NUMERICAL STABILITY FOR ANY SIZE TIME-STEP.
 - (B) STEADY-STATE FISSION POWER ITERATION FOR INITIAL EQUILIBRIUM CONDITIONS.
 - (C) ADJUSTABLE TIME-STEPS.
 - (D) PIECE-WISE LINEAR OR QUADRATIC TIME-DEPENDENT CROSS SECTIONS.
 - (E) PIECE-WISE LINEAR TIME-DEPENDENT INHOMOGENEOUS SOURCE TERM.
 - (F) FEEDBACK DUE TO THE TIME-DEPENDENT BUILDUP IN XENON.
8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED NOVEMBER 1967.
10. REFERENCES - K. F. HANSEN AND J. B. ANDREWS, NUMERICAL SOLUTION OF THE TIME DEPENDENT DIFFUSION EQUATION, TRANSACTIONS AMERICAN NUCLEAR SOCIETY, 9, 1966.
K. F. HANSEN AND S. JOHNSON, GAKIN - A ONE DIMENSIONAL MULTIGROUP KINETICS CODE, GAMC-7543, AUGUST 1967.
B. A. ANDREWS, NUMERICAL SOLUTION OF THE SPACE-DEPENDENT REACTOR KINETICS EQUATION, THESIS MIT, MAY 1967.
11. MACHINE REQUIREMENTS - 64K CCRE MEMORY WITH 2 TAPES (I/O)
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II, GAX27.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
S. R. JOHNSON
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (3792 CARDS)
SAMPLE PROBLEM (34 CARDS)
REFERENCE REPORT, GAMC-7543
17. CATEGORY - F
KEYWORDS - KINETICS, MULTIGROUP, 1-DIMENSIONAL, DIFFUSION EQUATIONS, FEEDBACK, SLABS, CYLINDERS, SPHERES

1. NAME OR DESIGNATION OF PROGRAM - BURNUP
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - GE635
3. NATURE OF PHYSICAL PROBLEM SOLVED - BURNUP CORRELATES HEAVY ELEMENT ISOTOPIC ANALYSIS WITH FISSION PRODUCT NEODYMIUM, URANIUM, AND PLUTONIUM CONCENTRATIONS IN AN IRRADIATED URANIUM FUEL FOR CALCULATION OF BURNUP (ATOM PER CENT FISSION AND MWD/MT). REACTOR PARAMETERS, INCLUDING EFFECTIVE NEUTRON ABSORPTION CROSS SECTIONS FOR ALL URANIUM AND PLUTONIUM ISOTOPES, CAPTURE-TO-FISSION RATIOS FOR U235, PU239, AND PU241, A TWO-GROUP DESCRIPTION OF THE NEUTRON SPECTRUM, THE AVERAGE NEUTRON TEMPERATURE, THE REACTOR FAST FISSION FACTOR, AND THE DISTRIBUTION OF THE SOURCES OF FISSION AMONG THE FISSIONABLE NUCLIDES ARE COMPUTED FROM THE EXPERIMENTAL DATA OBTAINED FROM THE MASS SPECTROMETRIC ANALYSIS OF URANIUM, PLUTONIUM, AND NEODYMIUM.
4. METHOD OF SOLUTION - AN ITERATIVE PROCESS IS UTILIZED WHEREBY A CORRECTION FOR TRACES OF URANIUM CONTAMINATION AS WELL AS CORRECTIONS BASED ON EXPOSURE AND DECAY FACTORS ARE OBTAINED ON AN INITIAL PASS THROUGH THE DATA. THE LAST ITERATION USING THE CORRECTIONS OBTAINED GIVES A FINAL VALUE FOR THE CALCULATIONS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - A MAXIMUM OF 50 SAMPLES CAN BE RUN IN ANY ONE CASE. ALTHOUGH A SINGLE SAMPLE CAN BE PROCESSED, THE AVERAGING PROCEDURES FOR NUCLEAR CONSTANTS ARE MOST EFFECTIVE WHEN 5 TO 50 SAMPLES FROM THE SAME FUEL LOAD ARE POOLED IN A SINGLE CASE.
6. TYPICAL RUNNING TIME - 2 TO 3 SECONDS PER SAMPLE ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
GE635 VERSION SUBMITTED DECEMBER 1967.
10. REFERENCES - B. F. RIDER AND C. P. RUIZ, DETERMINATION OF ATOM PERCENT FISSION IN URANIUM FUEL, PROGRESS IN NUCLEAR ENERGY SERIES IX, ANALYTICAL CHEMISTRY 3, PART 2, C. E. CROUTHAMER, ED., PERGAMON PRESS, 1962.
B. F. RIDER, J. PAUL PETERSON, JR., J. L. JAECH, C. P. RUIZ AND F. R. SMITH, ACCURATE NUCLEAR BURNUP ANALYSIS XV, GEAP-4950, SEPTEMBER 1965.
B. F. RIDER, C. P. RUIZ, J. P. PETERSON, JR., AND F. R. SMITH, BURNUP, A FORTRAN IV CODE FOR COMPUTING U AND PU FUEL BURNUP FROM U, PU, AND ND MASS SPECTROMETRIC MEASUREMENTS, GEAP-5355, OCTOBER 31, 1967.

10. REFERENCES (CONTINUED)

BURNUP SUBROUTINE DESCRIPTIONS INCLUDING -

BITA, L5T001, DECEMBER 3, 1964.

A. E. SCADDEN, ZERO, DIVISION OVERFLOW MODIFIER,
Q4T001.

NELS J. ANDERSON, JR., IDFTAP FUNCTION SUBPROGRAM
WRITE-UP, GE-NED NCTE, 1968.

11. MACHINE REQUIREMENTS - 32K MEMORY AND AN ON-LINE PRINTER

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
GECOS.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
A NON-STANDARD DATA LOADING ROUTINE (DTA) IS UTILIZED TO READ
INPUT CARDS. DTA IS FORTRAN IV COMPATIBLE AND READS AND LOADS
NUMERIC AND HOLLERITH (BCD) DATA PUNCHED FREE STYLE ON THE 80-
COLUMN CARD.

15. NAME AND ESTABLISHMENT OF AUTHORS -

B. F. RIDER, J. P. PETERSON, JR.,
F. R. SMITH, AND T. V. MCLAUGHLIN
VALLECITOS NUCLEONICS LABORATORY
GENERAL ELECTRIC COMPANY
PLEASANTON, CALIFORNIA

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECK (4754 CARDS)

SAMPLE PROBLEM (51 CARDS)

REFERENCE REPORT, GEAP-5355, AND BURNUP SUBROUTINE DESCRIPTIONS

17. CATEGORY - C

KEYWORDS - FISSION PRODUCTS, HEAVY ISOTOPES, DEPLETION, MEASURE-
MENTS, FUELS, URANIUM, NEODYMIUM, PLUTONIUM

1. NAME OR DESIGNATION OF PROGRAM - TDSN
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090,94
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE LINEAR, TIME-INDEPENDENT, BOLTZMANN EQUATION IS SOLVED FOR THE ENERGY, SPACE, AND ANGULAR DEPENDENCE OF THE NEUTRON DISTRIBUTION IN ONE-DIMENSIONAL SLABS, CYLINDERS OR SPHERES OR IN TWO-DIMENSIONAL X-Y OR R-Z GEOMETRY. FIXED SOURCE OR MULTIPLICATION FACTOR (ADJOINT OR FLUX) PROBLEMS ARE SOLVED SUBJECT TO VACUUM, PLANE REFLECTIVE, ISOTROPIC REFLECTIVE OR 180 DEGREE ROTATIONALLY SYMMETRIC BOUNDARY CONDITIONS. A COMPLETE ENERGY TRANSFER SCATTERING MATRIX IS ALLOWED FOR EACH LEGENDRE COMPONENT OF SCATTERING CROSS SECTION MATRICES THROUGH P1.
4. METHOD OF SOLUTION - ENERGY DEPENDENCE IS TREATED BY THE MULTI-GROUP APPROXIMATION AND ANGULAR DEPENDENCE BY A DISCRETE ORDINATES APPROXIMATION. WITHIN-GROUP SCATTERING AND UPSCATTERING ITERATION PROCESSES ARE ACCELERATED BY RE-NORMALIZATION PROCEDURES. INNER AND OUTER ITERATION ACCELERATION MAY BE DONE BY OVER-RELAXATION.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - 17,500 STORAGES IN A 32K MACHINE ARE AVAILABLE FOR VARIABLES IN ANY COMBINATION OF GROUPS, SPACE INTERVALS, ANGULAR QUADRATURE, ETC. THE ARRAY OF 17,500 STORAGES MAY BE INCREASED OR DECREASED BY RECOMPILING FOR A PARTICULAR MACHINE CONFIGURATION. A COUNTER KEEPS TRACK OF PROBLEM STORAGE REQUIREMENTS AND PRINTS THE NUMBER REQUIRED FOR EACH OVERLAY.
6. TYPICAL RUNNING TIME - RUNNING TIMES ARE FROM LESS THAN 1 MINUTE PER PROBLEM TO MORE THAN 1 HOUR PER OUTER ITERATION.
7. UNUSUAL FEATURES OF THE PROGRAM - SEVERAL CONVERGENCE TESTS AND INDICATORS ARE USED. TDSN USES EITHER DIAMOND OR STEP MODELS OR BOTH AT THE USER'S OPTION. PROBLEMS CAN BE RESTARTED FROM A BINARY CARD DUMP OR, WITH MINOR MODIFICATIONS, A TAPE RESTART. THREE FORMATS FOR CROSS SECTION INPUT ARE AVAILABLE - TDSN, DIFFUSION THEORY, AND DTF OR DDF FORMATS. OVER-RELAXATION FOR INNER AND OUTER ITERATION ACCELERATION MAY BE USED.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM7090 VERSION SUBMITTED DECEMBER 1967.
10. REFERENCES - C. E. BARBER, A FORTRAN IV TWO-DIMENSIONAL DISCRETE ANGULAR SEGMENTATION TRANSPORT PROGRAM, NASA TN D-3573, AUGUST 1966 AND MEMORANDUM OF CHANGES TO THE INPUT INSTRUCTIONS NASA TN D-3573.
11. MACHINE REQUIREMENTS - 32K MEMORY

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12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND MAP
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
DIRECT-COUPLED 7040-7094.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
A CLOCK READING MAP SUBROUTINE IS USED BUT MAY BE REPLACED BY
USERS EQUIVALENT OR A DUMMY. MAP SUBROUTINES ARE USED FOR RESTART
ONLY. THEY MAY BE ELIMINATED, REPLACED, OR CHANGED FOR TAPE
RESTART IF DESIRED.
15. NAME AND ESTABLISHMENT OF AUTHOR -
C. E. BARBER *
LEWIS RESEARCH CENTER
NATIONAL AERONAUTICS AND SPACE ADMINISTRATION
21000 BROOKPARK ROAD
CLEVELAND, OHIO 44135
* NO LONGER AT LEWIS - CONTACT WENDELL MAYO AT ABOVE ADDRESS.
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (5730 CARDS)
SAMPLE PROBLEM (100 CARDS)
REFERENCE REPORT AND MEMORANDUM
17. CATEGORY - C
KEYWORDS - 2-DIMENSIONAL, SN METHOD, X-Y, R-Z GEOMETRIES, MULTI-
GROUP, 1-DIMENSIONAL, SLABS, CYLINDERS, SPHERES, REAC-
TIVITY, FLUX DISTRIBUTION

1. NAME OR DESIGNATION OF PROGRAM - CINDER(M0102)
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - CINDER IS A FOUR-GROUP, ONE-POINT DEPLETION AND FISSION PRODUCT PROGRAM BASED ON THE EVALUATION OF A GENERAL ANALYTICAL SOLUTION OF NUCLIDES COUPLED IN ANY LINEAR SEQUENCE OF RADIOACTIVE DECAYS AND NEUTRON ABSORPTIONS IN A SPECIFIED NEUTRON FLUX SPECTRUM. THE DESIRED DEPLETION AND FISSION PRODUCT CHAINS AND ALL PHYSICAL DATA ARE SPECIFIED BY THE PROBLEM ORIGINATOR. THE PROGRAM COMPUTES INDIVIDUAL NUCLIDE NUMBER DENSITIES, ACTIVITIES, NINE ENERGY-GROUP DISINTEGRATION RATES, AND MACROSCOPIC AND BARNES/FISSION POISONS AT EACH TIME-STEP AS WELL AS SELECTED SUMMARIES OF THESE DATA.
4. METHOD OF SOLUTION - TIME-DEPENDENT VARIATIONS IN NUCLIDE CROSS SECTIONS AND NEUTRON FLUXES ARE APPROXIMATED BY A USER-SPECIFIED SEQUENTIAL SET OF VALUES WHICH ARE CONSIDERED CONSTANT DURING THE DURATION OF THE USER-SPECIFIED ASSOCIATED TIME-INCREMENTS. WHEN A NUCLIDE CONCENTRATION IS INDEPENDENT OF THE CONCENTRATION OF ANY OF ITS PROGENY, IT IS POSSIBLE TO RESOLVE THE COUPLINGS SO AS TO OBTAIN NUCLIDES FED BY A SINGLE PARENT. THESE CHAINS ARE REFERRED TO AS LINEAR.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE PROGRAM IS LIMITED TO 500 TOTAL NUCLIDES FORMED IN UP TO 240 CHAINS OF 20 OR FEWER NUCLIDES EACH. UP TO 10 NUCLIDES MAY ACT AS FISSION PRODUCT SOURCES, CONTRIBUTING TO POWER, AND AS MANY AS 99 TIME-STEPS OF ARBITRARY LENGTH ARE PERMITTED. ALL STABLE NUCLIDES MUST HAVE A CROSS SECTION IF ZERO POWER TIME-INCREMENTS ARE ANTICIPATED.
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM - DURING THE NUCLIDE CONCENTRATION CALCULATION THE PROGRAM CHECKS FOR ERRORS ORIGINATING FROM THE FINITE WORD LENGTH OF THE COMPUTER.
8. RELATED AND AUXILIARY PROGRAMS - A PREVIOUS VERSION OF CINDER EXISTED FOR THE PHILCO 2000.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC6600 VERSION SUBMITTED DECEMBER 1967.
10. REFERENCES - T. R. ENGLAND, TIME-DEPENDENT FISSION-PRODUCT THERMAL AND RESONANCE ABSORPTION CROSS SECTIONS, WAPD-TM-333, NOVEMBER 1962 AND ADDENDUM 1, JANUARY 1965.
T. R. ENGLAND, CINDER - A ONE POINT DEPLETION AND FISSION PRODUCT PROGRAM, WAPD-TM-334 (REVISED), AUGUST 1962, REVISED JUNE 1964.
J. A. DUGAN, CINDER (M0102) ADDITIONAL OPTIONS - MEMORANDUM 2, WAPC-L-(PA)-89 PRELIMINARY.

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11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
 - T. R. ENGLAND
 - WESTINGHOUSE ELECTRIC CORPORATION
 - BETTIS ATOMIC POWER LABORATORY
 - P. O. BOX 79
 - WEST MIFFLIN, PENNSYLVANIA 15122
16. MATERIAL AVAILABLE -
 - SOURCE DECK (6600-892 CARDS)
 - SAMPLE PROBLEM (6600-52 CARDS)
 - REFERENCE REPORTS, ADDENDUM, AND MEMORANDUM
17. CATEGORY - D
KEYWORDS - DEPLETION, RADIOACTIVITY, DECAY, FISSION PRODUCTS,
MULTIGROUP, POISON, ABSORPTION

1. NAME OR DESIGNATION OF PROGRAM - NAP
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7094
3. NATURE OF PHYSICAL PROBLEM SOLVED - NAP CALCULATES THE SPECTRUM AND SPATIAL DISTRIBUTION IN ONE DIMENSION OF ACTIVATION GAMMA RAYS FOLLOWING NEUTRON IRRADIATION.
4. METHOD OF SOLUTION -
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 20 SPATIAL REGIONS
 - 20 ISOTOPES PER REGION
 - 10 NCN - 1/V ISOTOPES
 - 100 SPATIAL POINTS
 - 200 TIME POINTS
 - 43 NEUTRON GROUPS
 - 20 GAMMA ENERGY GROUPSTHE CHAIN LENGTH IS LIMITED TO A MAXIMUM OF 5. ONLY (N,GAMMA), (N,ALPHA), (N,P), AND (N,2N) REACTIONS ARE CONSIDERED.
6. TYPICAL RUNNING TIME - A 23-ISOTOPE PROBLEM ON AN OPTICALLY THIN MEDIUM WAS RUN IN 4 MINUTES. THIS TIME WOULD BE INCREASED IF THE SN CALCULATION OF THE NEUTRON DISTRIBUTION OR THE CROSS SECTION CALCULATION OPTION HAD BEEN UTILIZED.
7. UNUSUAL FEATURES OF THE PROGRAM - NAP PERMITS AN ARBITRARY NEUTRON SOURCE SPECTRUM OR ANGULAR DISTRIBUTION TO BE INCIDENT UPON A MEDIUM WHICH CAN BE DESCRIBED IN ONE DIMENSION (SPHERICAL OR CARTESIAN COORDINATES). ANY ISOTOPIC COMPOSITION OF THE MEDIUM IS PERMISSIBLE. IF THE APPROPRIATE CROSS SECTION IS NOT IN THE LIBRARY, THE USER HAS THE OPTION OF USING A THEORETICAL ESTIMATE OF THE CROSS SECTION PROVIDED BY THE CODE. THE ADDED RUNNING TIME RESULTING FROM THE USE OF THIS OPTION HAS NOT BEEN EVALUATED. ONE OR TWO MINUTES PER REACTION CROSS SECTION FOR AN 8 ENERGY GROUP PROBLEM HAS BEEN ESTIMATED.
8. RELATED AND AUXILIARY PROGRAMS - TWO AUXILIARY ROUTINES ARE INCLUDED TO PROVIDE AN EDITED LIBRARY LISTING. XPREP PRINTS THE NAP CROSS SECTION LIBRARY, AND RLBP PRINTS THE NAP GAMMA RADIATION LIBRARY.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM7094 VERSION SUBMITTED JANUARY 1968.
10. REFERENCES - D. A. KLOPP, NAP CODE MANUAL, IITRI-A6088-21, VOL. I, MAY 14, 1964 - JANUARY 31, 1966, AND XLIB CORRECTIONS.
D. A. KLOPP, NAP PHYSICAL MODELS AND EXPERIMENTAL VALIDATION, IITRI-A6088-21, VOL. II, MAY 14, 1964 - JANUARY 31, 1966.

10. REFERENCES (CONTINUED)

D. A. KLOPP, NAP CROSS SECTION LIBRARY, IITRI-A6088-21, VOL. III, MAY 14, 1964 - JANUARY 31, 1966.

D. A. KLOPP, NAP GAMMA RADIATION LIBRARY, IITRI-A6088-21, VOL. IV, MAY 14, 1964 - JANUARY 31, 1966.

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
WHEN USING XPREP TO PRINT THE NAP CROSS SECTION LIBRARY, LOAD THE
BINARY LIBRARY TAPE ON UNIT 8. WHEN USING RLBP TO PRINT THE NAP
GAMMA RADIATION LIBRARY, LOAD THE BINARY LIBRARY TAPE ON UNIT 1C.

15. NAME AND ESTABLISHMENT OF AUTHORS -

D. A. KLOPP AND W. E. ZAGOTTA
IIT RESEARCH INSTITUTE
1C WEST 35TH STREET
CHICAGO, ILLINOIS 60616

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECKS (NAP 2577 CARDS, XPREP 146 CARDS, RLBP 61 CARDS)

BINARY DECK (NAP 1032 CARDS)

SAMPLE PROBLEM (NAP 24 CARDS)

LIBRARIES (NEUTRON CROSS SECTION-98 BINARY RECORDS, GAMMA RAY-
121 BINARY RECORDS)

REFERENCE REPORTS, VOLS. I AND II

17. CATEGORY - D

KEYWORDS - GAMMA RADIATION, 1-DIMENSIONAL, MULTIGROUP, TRANSPORT
THEORY, ISOTOPES, DECAY, DOSE RATES, NEUTRON ACTIVA-
TION

1. NAME OR DESIGNATION OF PROGRAM - WIREX
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090
3. NATURE OF PHYSICAL PROBLEM SOLVED - WIREX PRODUCES WIRING LISTS CONTAINING ALL THE INFORMATION NORMALLY FOUND ON DETAILED WIRING DRAWINGS. THESE LISTS ARE PRESENTED TO THE ELECTRICIANS AS A JOB BOOK CONTAINING INFORMATION IN THE ORDER IN WHICH IT IS REQUIRED FOR LOGICAL PROCEDURE THROUGH A WIRING JOB. WIREX LISTS WERE USED EXTENSIVELY DURING INSTALLATION FOR THE UHTREX PROJECT.
4. METHOD OF SOLUTION - THE GENERAL NOTES GIVE INSTRUCTIONS AND SPECIAL ROUTING DETAILS. THE MATERIAL NOTES GIVE QUANTITIES AND TYPES OF MATERIALS TO BE DRAWN OR ORDERED FROM THE SOURCE INDICATED. THE CABLE LENGTH SUMMARY INDICATES THE TYPES AND LENGTH OF CABLE TO BE PULLED. THE CABLE ROUTING LIST, CHANNEL LISTS AND CABINET WIRING LISTS GIVE ROOM, CHANNEL, OR CABINET DATA FOR THE ELECTRICIAN, DESCRIBE HOW THE WIRES ARE CONNECTED TO A DEVICE OR TRANSDUCER, AND DEFINE ALL THE CONNECTIONS IN ORDER OF ASCENDING STRIP AND TERMINAL NUMBER FOR A CABINET OR STRIP.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 2000 PATH CARD PAIRS PER JOB
 - 49 ROOMS PER JOB
 - 100 CABINETS PER JOB
 - 350 CABLES PER JOB
 - 999 GENERAL NOTES CARDS PER JOB
 - 999 MATERIAL NOTES CARDS PER JOB
 - 100 STRIPS PER CABINET
 - 1000 PATHS AND JUMPERS PER STRIP
 - 1000 JUMPERS PER CABINET
6. TYPICAL RUNNING TIME - THE MOST COMPLEX PROBLEM RUN REQUIRED ABOUT 4 MINUTES AND PRODUCED 150 PAGES OF OUTPUT.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM7090 VERSION SUBMITTED JANUARY 1968.
10. REFERENCES - K. H. DUERRE AND J. R. CASE, WIREX, A FORTRAN COMPUTER PROGRAM FOR PRODUCING WIRE LISTS, LA-3807, DECEMBER 1967.
R. K. NELSON, DISK-TAPE VERSION OF FORTRAN-II (IOB), LASL NOTE IB-IOBD, JULY 17, 1963.
11. MACHINE REQUIREMENTS - WIREX USES 450 DRUM TRACKS, 2200 WORDS OF DISC STORAGE AND NEARLY 32K OF CORE MEMORY.
12. PROGRAMMING LANGUAGE USED - FORTRAN II

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
RANDOM TRANSFER OF DATA BETWEEN CORE AND AUXILIARY STORAGE WAS
MADE BY A DISC-TAPE VERSION OF FORTRAN II IOB WHICH ALLOWS READING
AND WRITING ON DISC OR DRUM BY READ TAPE N, LIST AND WRITE TAPE N,
LIST STATEMENTS WHERE N MAY HAVE BEEN CALCULATED.
15. NAME AND ESTABLISHMENT OF AUTHOR -
K. H. DUERRE
LOS ALAMOS SCIENTIFIC LABORATORY
P. O. BOX 1663
LOS ALAMOS, NEW MEXICO
16. MATERIAL AVAILABLE -
SOURCE DECK (981 CARDS)
SAMPLE PROBLEM (5 CARDS)
REFERENCE REPORT AND NOTE
17. CATEGORY - I
KEYWORDS - DESIGN, UHTREX REACTORS

1. NAME OR DESIGNATION OF PROGRAM - GAFGAR/P3T/PROC/TAPCOP
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE PROBLEM IS TO OBTAIN VERY DETAILED NEUTRON FLUX AND CURRENT DISTRIBUTIONS AS FUNCTIONS OF ENERGY CONSIDERING EXPLICITLY THE POSSIBLE OVERLAP EFFECTS BETWEEN RESONANCES OF A RESONANCE ABSORBER AND OF MIXTURES OF RESONANCE ABSORBERS AND TO USE THESE DISTRIBUTIONS TO PREPARE GROUP-AVERAGED CROSS SECTIONS AND TRANSFER ARRAYS FOR USE IN FAST REACTOR ANALYSES.
4. METHOD OF SOLUTION - THE B1 APPROXIMATION TO THE BOLTZMANN EQUATION IS SOLVED FOR THE NEUTRON FLUX AND CURRENT SPECTRA. THE GAF SECTION USES A LARGE NUMBER OF FINE ENERGY GROUPS TO REPRESENT THE CROSS SECTIONS ABOVE THE RESOLVED RESONANCE REGION. THE GAR SECTION USES A VERY LARGE NUMBER OF DISCRETE VELOCITIES IN THE RESONANCE REGION ALONG WITH SLOWING DOWN SOURCES FROM THE GAF SECTION TO TREAT LEAKAGE-DEPENDENT RESONANCE ABSORPTION MORE RIGOROUSLY THAN PREVIOUS CODES.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE MAXIMUM OF 1740 FINE GROUPS IN THE GAF SECTION MAY BE SUBDIVIDED INTO 10 OR LESS SETS OF FINE GROUPS HAVING A CONSTANT LETHARGY WIDTH. THE GAR SECTION REQUIRES A UNIFORM SPACING BETWEEN THE MAXIMUM OF 14,700 DISCRETE VELOCITIES. THE MAXIMUM NUMBER OF NUCLIDES IN A PROBLEM IS 20, AND THE MAXIMUM TOTAL NUMBER OF BROAD GROUPS (OUTPUT FOR NUCLEAR DESIGN CALCULATIONS) IS 99 WITH A MAXIMUM OF 20 BROAD GROUPS IN THE GAR SECTION.
6. TYPICAL RUNNING TIME - 4.6 MINUTES (1.1 MINUTES FOR GAF SECTION AND 3.5 MINUTES FOR GAR SECTION) ARE REQUIRED FOR A TEST PROBLEM INVOLVING HYDROGEN, CARBON, OXYGEN, PU239, AND PU240 WITH 731 FINE GROUPS ABOVE 10 KEV IN THE GAF SECTION ($\Delta\tau=0.01$), 12,367 VELOCITIES BELOW 10 KEV IN THE GAR SECTION ($\Delta v=110$ M/SEC), AND 31 BROAD GROUP OUTPUT (11 IN GAF SECTION AND 20 IN GAR SECTION).
7. UNUSUAL FEATURES OF THE PROGRAM - THE GAF SECTION ALLOWS DOWNSCATTER BY ANY PROCESS (ANISOTROPIC ELASTIC SCATTER, INELASTIC SCATTER, (N,2N) PROCESSES, ETC.) FROM ANY FINE GROUP TO ANY LOWER ENERGY FINE GROUP BECAUSE THE FULL P0 AND P1 TRANSFER ARRAYS ARE USED FOR ALL MATERIALS EXCEPT HYDROGEN WHICH IS ASSUMED TO HAVE ONLY ISOTROPIC IN THE CENTER-OF-MASS SYSTEM ELASTIC SCATTER. DATA COMPRESSION TECHNIQUES ARE USED TO REDUCE THE SIZE OF THE TRANSFER ARRAYS. THE GAR SECTION USES A RECURSIVE SCHEME FOR THE CALCULATION OF THE P0 AND P1 SLOWING DOWN SOURCE TERMS WHICH DECREASES THE RUNNING TIME FOR TYPICAL PROBLEMS BY A FACTOR OF 3 TO 4 WHILE STILL ALLOWING ISOTROPIC IN THE CENTER-OF-MASS SYSTEM ELASTIC SCATTER TO ANY LOWER VELOCITY TO BE TREATED RIGOROUSLY. EITHER POSITIVE OR NEGATIVE BUCKLING CAN BE SPECIFIED FOR EACH BROAD GROUP IN EACH SECTION.

8. RELATED AND AUXILIARY PROGRAMS - CROSS SECTION INPUT FOR THE GAFGAR CODE IS PREPARED BY THE GAND CODE FROM ENDF/B FORMAT NUCLEAR DATA FILE TAPES. DATA ON THE GAFGAR CROSS SECTION TAPES MAY BE PRINTED WITH THE P3T CODE. PROC IS USED TO COPY SELECTED PROBLEMS FROM 1 OR 2 GAF AND/OR GAR PROBLEM TAPES OR TO LIST THE PROBLEM TAPES. TAPCOP IS USED TO COPY GAFGAR (PACKED RECORD) DATA TAPES.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED JANUARY 1968, REPLACED BY
MODIFIED VERSION AUGUST 1968.
10. REFERENCES - D. R. MATHEWS AND R. J. ARCHIBALD, GAF, A COMPUTER PROGRAM FOR CALCULATION OF NEUTRON SPECTRA AND AVERAGE CROSS SECTIONS IN THE HIGH ENERGY RANGE, GA-7169, JANUARY 20, 1967.
C. A. STEVENS AND R. J. ARCHIBALD, GAR, A COMPUTER PROGRAM FOR EVALUATING LEAKAGE DEPENDENT RESONANCE ABSORPTION, GA-6952, MARCH 16, 1966.
R. J. ARCHIBALD AND D. R. MATHEWS, THE GAF/GAR/GAND FAST REACTOR CROSS SECTION PREPARATION SYSTEM, GAFGAR - A PROGRAM FOR THE CALCULATION OF NEUTRON SPECTRA AND GROUP-AVERAGED CROSS SECTIONS, GA-7542, VOL. I, JANUARY 22, 1968, AND ERRATA, MAY 1968.
11. MACHINE REQUIREMENTS - 52K OF AVAILABLE FAST MEMORY AND 5 MAGNETIC TAPE UNITS PLUS 5 SCRATCH FILES (DRUM FILES ARE USED ON THE UNIVAC1108) TOTALING ABOUT 600K IN CAPACITY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II, GAX33A.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - GAFGAR IS OVERLAYED IN THREE SEGMENTS CONSISTING OF THE GAF SECTION, THE GAR SECTION, AND A COMBINING AND PUNCHING SECTION. THE UNIVAC1107/1108 FORTRAN IV NTRAN I/O ROUTINES ARE USED TO BUFFER THE I/O IN PARALLEL WITH COMPUTING AND TO REDUCE THE NUMBER OF RECORD GAPS ON THE DATA TAPES.
15. NAME AND ESTABLISHMENT OF AUTHORS -
R. J. ARCHIBALD, D. R. MATHEWS, R. T. SHANSTROM,
C. A. STEVENS, AND R. M. WAGNER
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SAN DIEGO, CALIFORNIA 92112
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (GAFGAR 4702 CARDS, P3T 328 CARDS, PROC 197 CARDS,
TAPCOP 144 CARDS)
SAMPLE PROBLEM INPUT AND OUTPUT (242 CARDS)
REFERENCE REPORTS AND ERRATA

17. CATEGORY - B

KEYWORDS - FAST REACTORS, MULTIGROUP, CROSS SECTIONS, NEUTRONS
SPECTRA, RESONANCE ABSORPTION, FLUX DISTRIBUTION, CUR-
RENTS DISTRIBUTION, P3T CODES, PROC CODES, TAPCOP CODES

REACTOR CODE ABSTRACT 317

6/68

1. NAME OR DESIGNATION OF PROGRAM - GAPOTKIN
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - GAPOTKIN IS A POINT KINETICS CODE THAT SOLVES THE SPACE-INDEPENDENT KINETICS EQUATIONS FOR A VERY GENERAL FORM OF THE REACTIVITY FUNCTION.
4. METHOD OF SOLUTION - THE FORM OF THE KINETICS EQUATIONS USED IS THE REACTIVITY-LIFETIME FORMULATION. THE REACTIVITY IS SPECIFIED IN TERMS OF STEP, LINEAR, AND QUADRATIC COEFFICIENTS PROPORTIONATE TO THE REACTOR POWER AND/OR TOTAL ENERGY RELEASE. THE NUMERICAL METHOD FOR THE INTEGRATION OF THE KINETICS EQUATIONS IS DISCUSSED IN DETAIL BY HANSEN, KOEN, AND LITTLE (SEE REFERENCE 3). THE OPERATING LIFE OF THE SYSTEM IS DIVIDED INTO VARIOUS TIME ZONES, EACH WITH ITS OWN SET OF PARAMETERS, IN ORDER TO TREAT SEQUENCES OF EVENTS FOR WHICH THE SYSTEM PROPERTIES CHANGE.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - GAPOTKIN WILL HANDLE UP TO 20 TIME ZONES AND WILL ALLOW FOR 50 POWER-DEPENDENT OR ENERGY-DEPENDENT REACTIVITY COEFFICIENTS PER TIME ZONE IF TABLES ARE USED. GENERAL DATA ARE STORED FOR UP TO 500 TIME-STEPS AND TIME DELAY DATA ARE STORED FOR 10,000 TIME-STEPS.
6. TYPICAL RUNNING TIME - FOR A PROBLEM WITH TWO TIME ZONES, THE FIRST WITH A 60-SECOND MAXIMUM OPERATING TIME AND NO TIME DELAY, THE SECOND CONTINUING TO 160 SECONDS AND WITH NO TIME DELAY, THE RUNNING TIME IS ABOUT 20 SECONDS. WITH DELAY TIMES FOR ZONE ONE OF 0.1 SECOND AND 0.4 SECOND, THE RUNNING TIMES ARE APPROXIMATELY 3.5 MINUTES AND 5 MINUTES, RESPECTIVELY.
7. UNUSUAL FEATURES OF THE PROGRAM - A UNIQUE FEATURE OF THE ENERGY FEEDBACK IS THE OPTION OF INCLUDING TIME DELAYS IN THE REACTIVITY. THERE IS ALSO PROVISION FOR INCLUDING TIME-VARYING EXTERNAL SOURCES. THE OVERALL VERSATILITY OF THE CODE ALLOWS FOR THE HANDLING OF COMPLEX SEQUENCES OF EVENTS SUCH AS A SUDDEN LOSS OF COOLANT FOLLOWED BY A CONTROL ROD DROP AND DELAYED REACTIVITY INSERTIONS DUE TO ENERGY BUILDUP. BUILT-IN DATA FOR U233, U235, AND PU239 ARE FROM ANL-5800 (REFERENCE 1).
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED JANUARY 1968.
10. REFERENCES - K. F. HANSEN AND P. K. KOCH, GAPOTKIN, A POINT KINETICS CODE FOR THE UNIVAC 1108, GA-8204, OCTOBER 25, 1967.
G. R. KEEPIN, PHYSICS OF NUCLEAR KINETICS, ADDISON-WESLEY PUBLISHING COMPANY, INC., READING, MASSACHUSETTS, 1965.

10. REFERENCES (CONTINUED)

K. F. HANSEN, B. V. KOEN, AND W. W. LITTLE, JR.,
STABLE NUMERICAL SOLUTION OF THE REACTOR KINETICS EQUATIONS,
NUCLEAR SCIENCE AND ENGINEERING, VOL. 22, NO. 1, P. 51, 1965.
REACTOR PHYSICS CONSTANTS, SECOND EDITION, ANL-5800,
1963.

11. MACHINE REQUIREMENTS - GAPOTKIN REQUIRES 4100 STORAGE LOCATIONS. THE DATA STORAGE REQUIREMENT IS ROUGHLY 33,000 LOCATIONS. TWO I/O DEVICES ARE REQUIRED FOR READING AND PRINTING AND ONE ADDITIONAL I/O DEVICE IS NECESSARY IF THE PROGRAM IS LOADED FROM TAPE.
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II, GAX29.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE DIMENSIONS OF MOST OF THE ARRAYS ARE HANDLED WITH PARAMETER STATEMENTS. ADAPTATION TO A PARTICULAR MACHINE WITH A FORTRAN IV COMPILER CAN BE MADE EASILY BY EITHER ADJUSTING THE PARAMETERS TO ALLOW FOR AVAILABLE MEMORY OR BY ELIMINATING THE PARAMETERS AND USING EXPLICIT DIMENSIONS.
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE -
SOURCE DECK (1319 CARDS)
SAMPLE PROBLEM (28 CARDS)
REFERENCE REPORT, GA-8204
17. CATEGORY - E
KEYWORDS - KINETICS, SPACE-INDEPENDENT, FEEDBACK, REACTIVITY

1. NAME OR DESIGNATION OF PROGRAM - FEVER7
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - FEVER7 PERFORMS A MULTIGROUP, 1-DIMENSIONAL DEPLETION CALCULATION WHICH ALLOWS FUEL IN VARIOUS STAGES OF IRRADIATION TO BE HOMOGENIZED INTO THE SAME REGION FOR PURPOSES OF THE DIFFUSION CALCULATION BUT FOLLOWS THE DEPLETION OF EACH OF THE SUBREGIONS SEPARATELY. THE CALCULATION MAY BE INTERRUPTED PERIODICALLY FOR REFUELING ONE OR MORE REGIONS. RECYCLING IS OPTIONAL AND THERE IS NO LIMIT TO THE NUMBER OF REFUELINGS WHICH MAY BE PERFORMED. A CONTROL POISON SEARCH IS AVAILABLE AND CONCENTRATION DEPENDENT SELF-SHIELDING FACTORS MAY BE APPLIED TO A NUMBER OF LUMPED POISONS.
4. METHOD OF SOLUTION - FLUX DISTRIBUTION AND MULTIPLICATION FACTORS ARE COMPUTED BY MEANS OF A MULTIGROUP DIFFUSION ROUTINE. THE DEPLETION CALCULATIONS ARE PERFORMED ASSUMING THAT THE FLUX AMPLITUDE, SPECTRUM AND SPATIAL DISTRIBUTION REMAIN CONSTANT FOR SHORT TIME-STEPS. THE FLUX IN THE THERMAL GROUPS IS RE-NORMALIZED TO THE POWER PRIOR TO EACH SHORT TIME-STEP AND THE SPECTRUM IS RECALCULATED AFTER A SPECIFIED SERIES OF SHORT TIME-STEPS. FOR THE BURNUP, NUCLIDES BELONGING TO EACH SUBREGION REACT TO THE AVERAGE FLUXES PRESENT IN THE REGION TO WHICH THE SUBREGION IS ASSIGNED. THE DIFFUSION CALCULATION IS PERFORMED USING GROUP CONSTANTS WHICH ARE VOLUME-WEIGHTED PROPERTIES OF ONE OR MORE SUBREGIONS HOMOGENIZED INTO A DIFFUSION REGION. THE MULTIGROUP 1-DIMENSIONAL DIFFUSION ROUTINE DCALC, INCORPORATED IN FEVER7, IS BASED ON A DIRECT MARCH OUT TECHNIQUE WITH PERIODIC RECONDITIONING. THE CONVENTIONAL FISSION SOURCE ITERATION HAS BEEN REPLACED BY WIELANDT ITERATIONS, SOLVING ALL GROUP FLUXES SIMULTANEOUSLY.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 7 ENERGY GROUPS
 - 151 MESH POINTS
 - 20 REGIONS
 - 60 SUBREGIONS
 - 45 NUCLIDES PER SUBREGION
 - FIXED SETS OF HEAVY NUCLIDES
6. TYPICAL RUNNING TIME - 10 SECONDS PER LARGE TIME-STEP ARE REQUIRED FOR A PROBLEM WITH 3 FAST GROUPS, 4 THERMAL GROUPS, AND 151 MESH POINTS.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - THIS CODE IS AN EXTENSION OF THE RELOAD FEVER PROGRAM (ACC ABSTRACT 221), FROM 4 TO 7 NEUTRON ENERGY GROUPS UTILIZING THE DCALC DIFFUSION EQUATION ROUTINE.

9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED JANUARY 1968.
10. REFERENCES - F. TODT, RELOAD FEVER, A ONE-DIMENSIONAL, FEW GROUP DEPLETION PROGRAM FOR REACTOR FUEL CYCLE ANALYSIS, GA-6612, SEPTEMBER 18, 1965.
M. R. WAGNER, DCALC, A SUBROUTINE FOR ONE-DIMENSIONAL NEUTRON DIFFUSION THEORY CALCULATIONS, GA-8286, OCTOBER 17, 1967.
11. MACHINE REQUIREMENTS - 65K UNIVAC1108 OR 1107 AND 3 TAPES (INPUT, OUTPUT, SYSTEM)
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (5941 CARDS)
SAMPLE PROBLEM (786 CARDS)
REFERENCE REPORTS
17. CATEGORY - C
KEYWORDS - MULTIGROUP, 1-DIMENSIONAL, DEPLETION, DIFFUSION EQUATIONS, REACTIVITY, FLUX DISTRIBUTION, POISON SEARCHES

1. NAME OR DESIGNATION OF PROGRAM - GASP7
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - GASP7 CALCULATES THE ONE-DIMENSIONAL DISTRIBUTION OF FISSILE AND FERTILE MATERIALS IN A NUCLEAR REACTOR WHICH WILL YIELD ANY DESIRED POWER DISTRIBUTION DURING THE BURNUP HISTORY OF THE REACTOR CORE. A POISON AND POISON DISTRIBUTION SEARCH FOR A DESIRED MULTIPLICATION AND MINIMUM POWER DISTRIBUTION PERTURBATION CAN ALSO BE PERFORMED.
4. METHOD OF SOLUTION - A MULTIGROUP, ONE-DIMENSIONAL DIFFUSION THEORY CALCULATION IS PERFORMED WITH AN ASSUMED FISSILE AND FERTILE FUEL DISTRIBUTION. AN ITERATIVE SCHEME IS EMPLOYED TO ADJUST THE FISSILE DISTRIBUTION TO ACHIEVE THE DESIRED POWER DISTRIBUTION AND TO ADJUST THE FERTILE DISTRIBUTION TO ACHIEVE A BURNUP INDEPENDENT POWER DISTRIBUTION. THE POISON AND POISON DISTRIBUTION SEARCH ARE PERFORMED BY AN ITERATIVE SCHEME WHICH MAINTAINS THE SAME DISTRIBUTION AS IN THE UNPOISONED CORE. THE ONE-DIMENSIONAL DIFFUSION ROUTINE DCALC, INCORPORATED IN GASP7, IS BASED ON A DIRECT MARCH OUT TECHNIQUE WITH PERIODIC RECONDITIONING, SOLVING FOR ALL THERMAL GROUP FLUXES SIMULTANEOUSLY.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 7 ENERGY GROUPS
 - 4 THERMAL NEUTRON ENERGY GROUPS
 - 151 MESH POINTS
 - 20 REGIONS
 - 45 DIFFERENT MATERIALS IN EACH REGION
 - 80 DIFFERENT CROSS SECTION SETSDOWNSCATTERING FROM THE FAST GROUPS INTO THE NEXT LOWER GROUP ONLY IS PERMITTED. THE GEOMETRY ASSUMED IS CYLINDRICAL, AND EITHER RADIAL OR AXIAL FLUX DISTRIBUTIONS MAY BE CALCULATED.
6. TYPICAL RUNNING TIME - 1 TO 2 MINUTES ARE REQUIRED FOR A 7-GROUP CASE.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - GASP7 IS AN EXTENSION OF GASP2 (ACC ABSTRACT 220) FROM 4 TO 7 NEUTRON ENERGY GROUPS, UTILIZING THE DCALC DIFFUSION EQUATION ROUTINE.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED JANUARY 1968.
10. REFERENCES - W. A. SIMON AND K. R. VAN HOWE, GASP-II, A COMPUTER CODE FOR BURNUP-INDEPENDENT POWER SHAPING BY FUEL ZONING INCORPORATING A POISON SEARCH, GA-6689, OCTOBER 11, 1965.
M. R. WAGNER, DCALC, A SUBROUTINE FOR ONE-DIMENSIONAL NEUTRON DIFFUSION THEORY CALCULATIONS, GA-8286, OCTOBER 17, 1967.

11. MACHINE REQUIREMENTS - 65K UNIVAC1108 OR 1107 AND 3 TAPES (INPUT, OUTPUT, SYSTEM)
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (4918 CARDS)
SAMPLE PROBLEM (404 CARDS)
REFERENCE REPORTS
17. CATEGORY - C
KEYWORDS - MULTIGROUP, ONE-DIMENSIONAL, DIFFUSION EQUATIONS, POWER DISTRIBUTION SEARCHES, POISON SEARCHES

1. NAME OR DESIGNATION OF PROGRAM - TEMCO7
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - TEMCO7 COMPUTES REACTOR TEMPERATURE COEFFICIENTS.
4. METHOD OF SOLUTION - GIVEN CROSS SECTIONS FOR EACH TEMPERATURE TO BE CONSIDERED, THE CODE PERFORMS A SERIES OF ONE-DIMENSIONAL DIFFUSION CALCULATIONS AND THEN USES THE RESULTS TO GENERATE THE TEMPERATURE COEFFICIENT. THE 1-DIMENSIONAL DIFFUSION SUBROUTINE DCALC IS BASED ON A DIRECT MARCH OUT TECHNIQUE WITH PERIODIC RECONDITIONING, SOLVING FOR ALL THERMAL GROUP FLUXES SIMULTANEOUSLY.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 25 TEMPERATURES
 - 3 FAST ENERGY GROUPS
 - 1-4 THERMAL ENERGY GROUPS
 - 20 REGIONS
 - 151 MESH POINTS
 - 60 SUBREGIONS
 - 40 NUCLIDES PER SUBREGION
 - 80 CROSS SECTION SETS PER TEMPERATURE
6. TYPICAL RUNNING TIME - A TIME ESTIMATE FOR A 7 ENERGY GROUP CASE IS 5 TO 10 SECONDS PER TEMPERATURE.
7. UNUSUAL FEATURES OF THE PROGRAM - TEMCO7 COMPUTES TEMPERATURE DERIVATIVES OF THE FOUR FACTORS η , ϵ , ρ , β , AND THE CONTRIBUTIONS OF EACH NUCLIDE TO THE TEMPERATURE COEFFICIENT.
8. RELATED AND AUXILIARY PROGRAMS - TEMCO7 IS AN EXTENSION OF TEMCO (ACC ABSTRACT 225) FROM 4 TO 7 NEUTRON ENERGY GROUPS UTILIZING THE DCALC DIFFUSION EQUATION ROUTINE.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED JANUARY 1968.
10. REFERENCES - F. W. TODT, TEMCO, A FORTRAN PROGRAM FOR COMPUTING TEMPERATURE COEFFICIENTS, GA-6734, DECEMBER 28, 1965.
M. R. WAGNER, DCALC, A SUBROUTINE FOR ONE-DIMENSIONAL NEUTRON DIFFUSION THEORY CALCULATIONS, GA-8286, OCTOBER 17, 1967.
11. MACHINE REQUIREMENTS - 65K UNIVAC1108 OR 1107 AND 3 TAPES (INPUT, OUTPUT, SYSTEM)
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (3705 CARDS)
SAMPLE PROBLEM (2643 CARDS)
REFERENCE REPORTS
17. CATEGORY - C
KEYWORDS - TEMPERATURE COEFFICIENTS, MULTIGROUP, 1-DIMENSIONAL,
DIFFUSION EQUATIONS, FLUX DISTRIBUTION, REACTIVITY,
RESONANCE ESCAPE PROBABILITY, THERMAL UTILIZATION,
FAST FISSION FACTOR

REACTOR CODE ABSTRACT 321

6/68

1. NAME OR DESIGNATION OF PROGRAM - EXPALS
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7094
3. NATURE OF PHYSICAL PROBLEM SOLVED - THIS PROGRAM FITS BY LEAST SQUARES A FUNCTION WHICH IS A LINEAR COMBINATION OF REAL EXPONENTIAL DECAY FUNCTIONS. THE FUNCTION IS
$$Y(K) = \text{SUMMATION OVER } J \text{ OF } A(J) * \text{EXP}(-\text{LAMBDA}(J) * K).$$
VALUES OF THE INDEPENDENT VARIABLE (K) AND THE DEPENDENT VARIABLE Y(K) ARE SPECIFIED AS INPUT DATA. WEIGHTS MAY BE SPECIFIED AS INPUT INFORMATION OR SET BY THE PROGRAM ($W(K) = 1/Y(K)$).
4. METHOD OF SOLUTION - THE PRONY-HOUSEHOLDER ITERATION METHOD IS USED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM7030 VERSION SUBMITTED JANUARY 1968, DELETED AUGUST 1968.
IBM7094 VERSION SUBMITTED AUGUST 1968.
10. REFERENCES - C. DOUGLAS GARDNER, EXPALS - FORTRAN CODE FOR EXPONENTIAL APPROXIMATION BY LEAST SQUARES, UCRL-14541, DECEMBER 2, 1965.
EXAMPLES OF THE USE OF EXPALS, UCRL NOTE, 1968.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN II
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - THE MONITOR OPERATING SYSTEM IS USED ON THE IBM7030.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE -
SOURCE DECK (766 CARDS)
SAMPLE PROBLEM (43 CARDS)
REFERENCE REPORT AND NOTE
17. CATEGORY - P
KEYWORDS - LEAST SQUARES, RADIOACTIVITY, DECAY

REACTOR CODE ABSTRACT 322

6/68

1. NAME OR DESIGNATION OF PROGRAM - ISOSEARCH
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360, CDC1604
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE PROGRAM WAS DEVELOPED TO CALCULATE THE UNKNOWN REACTION CROSS SECTION, FLUX VALUE, OR PRODUCT ACTIVITY IN AN ISOTOPE-PRODUCTION SCHEME CONSISTING OF TWO OR THREE NUCLIDES.
4. METHOD OF SOLUTION - THE PROGRAM IS BASED ON SOLUTIONS TO THE BATEMAN EQUATIONS FOR REACTION AND DECAY CHAINS. THE SOLUTIONS FOR A CHAIN CONTAINING TWO OR THREE NUCLIDES ARE COMBINED TO FORM A GENERALIZED SOLUTION APPLICABLE TO MOST SIMPLE ISOTOPE-PRODUCTION OR FLUX-DETERMINATION PROBLEMS. A SEARCHING PROCEDURE IS USED WHICH STARTS WITH GIVEN MAXIMUM AND MINIMUM ESTIMATES FOR THE UNKNOWN VARIABLE AND CONVERGES TO A VALUE THAT FITS THE INPUT DATA GIVEN FOR THE KNOWN VARIABLES.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM - THE INPUT DATA ARE SET UP TO FACILITATE THE INVESTIGATION OF DIFFERENT VALUES OF IRRADIATION TIME, DISINTEGRATIONS PER SECOND, OR NEUTRON FLUX FOR THE CHAIN CONSIDERED.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM360 VERSION SUBMITTED JANUARY 1968.
CDC1604 VERSION SUBMITTED JANUARY 1968.
10. REFERENCES - C. W. FRIEND, R. E. LEWIS, AND A. R. JENKINS, ISOSEARCH - COMPUTER PROGRAM FOR CALCULATING UNKNOWN VARIABLES IN REACTION AND DECAY SCHEMES, CRNL-3905, JANUARY 1967.
R. P. RANNIE, CHARACTER MANIPULATION SUBROUTINES FOR THE IBM/360, CRNL MEMO NO. 2/360.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGES USED - FORTRAN IV (IBM360) AND FORTRAN 63 (CDC1604)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

C. W. FRIEND AND R. E. LEWIS, ISOTOPES DIVISION
A. R. JENKINS, MATHEMATICS DIVISION
OAK RIDGE NATIONAL LABORATORY
P. O. BOX X
OAK RIDGE, TENNESSEE 37831

16. MATERIAL AVAILABLE -

SOURCE DECKS (360-610 CARDS, 1604-300 CARDS)
SAMPLE PROBLEMS (360-30 CARDS, 1604-30 CARDS)
REFERENCE REPORT AND MEMO (360)

17. CATEGORY - C

KEYWORDS - ISOTOPES PRODUCTION, DECAY, CAPTURE, CROSS SECTIONS,
FLUX

REACTOR CODE ABSTRACT 323

6/68

1. NAME OR DESIGNATION OF PROGRAM - MUFFLE
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090
3. NATURE OF PHYSICAL PROBLEM SOLVED - THIS PROGRAM COMPUTES THE NEUTRON CROSS SECTIONS FOR A FISSIONABLE NUCLIDE IN WHICH ONE TO THREE REACTION CHANNELS ARE OPEN FOR THE FISSION PROCESS. PROVISION IS MADE FOR TWO INDEPENDENT SPIN STATES WITH INTERFERING LEVELS AS WELL AS A SET OF NON-INTERFERING LEVELS. THE CROSS SECTIONS ARE INTEGRATED OVER A SPECIFIED GROUP STRUCTURE TO YIELD RESONANCE INTEGRALS.
4. METHOD OF SOLUTION - THIS MULTILEVEL ANALYSIS PROGRAM FOLLOWS THE THEORY DEVELOPED BY WIGNER AND EISENBUD, LANE AND THOMAS, AND VOGT. THE CROSS SECTIONS ARE DEFINED BY SPECIFYING FOR EACH LEVEL - THE LEVEL ENERGY, THE NEUTRON WIDTH, THE TOTAL CAPTURE WIDTH, THE TOTAL FISSION WIDTH, AND THE ANGLES SPECIFYING THE DIRECTION OF THE FISSION VECTOR.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
30 LEVELS FOR EACH SPIN
100 MAJOR ENERGY INTERVALS
500 SUBINTERVALS OF EQUAL LETHARGY WIDTH IN A MAJOR INTERVAL
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM - A $1/V$ CONTRIBUTION MAY BE ADDED TO THE CAPTURE AND ABSORPTION CROSS SECTIONS. A SET OF NON-INTERFERING LEVELS MAY BE INCLUDED, THESE ENTERING ONLY ON THE DIAGONAL OF THE INVERSE LEVEL MATRIX. THE POTENTIAL SCATTERING PHASE SHIFT IS COMPUTED APPROPRIATE TO S-WAVE NEUTRONS. IF REQUESTED, A TAPE IS WRITTEN FOR PLOTTING ON THE BENSON-LEHNER ELECTROPLOTTER.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM7090 VERSION SUBMITTED JANUARY 1968.
10. REFERENCE - C. A. PRESKITT, MUFFLE - A CODE FOR THE MULTILEVEL ANALYSIS OF NEUTRON CROSS SECTIONS, ORNL-TM-1180, APRIL 1965.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - IBSYS.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -

C. A. PRESKITT
OAK RIDGE NATIONAL LABORATORY
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OAK RIDGE, TENNESSEE 37830

16. MATERIAL AVAILABLE -

SOURCE DECK (680 CARDS)
BINARY DECK (471 CARDS)
SAMPLE PROBLEM (15 CARDS)
REFERENCE REPORT

17. CATEGORY - A

KEYWORDS - MULTILEVEL, ABSORPTION, CAPTURE, FISSION, SCATTERING,
CROSS SECTIONS, RESONANCE INTEGRALS

REACTOR CODE ABSTRACT 324

6/68

1. NAME OR DESIGNATION OF PROGRAM - FRANTIC
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090
3. NATURE OF PHYSICAL PROBLEM SOLVED - FRANTIC IS DESIGNED TO PROCESS RAW COUNTING DATA AND TO FIT IN THE LEAST SQUARES SENSE THESE DATA TO THE MULTIPLE EXPONENTIAL GROWTH AND DECAY EQUATIONS. THE PROGRAM CAN BE USED FOR SUMS OF EXPONENTIALS WITH POSITIVE, NEGATIVE, OR ZERO EXPONENTS AND POSITIVE OR NEGATIVE COEFFICIENTS.
4. METHOD OF SOLUTION - IN ORDER TO USE THE LEAST SQUARES METHOD THE EQUATIONS MUST BE MADE LINEAR BY EXPANDING EACH EXPRESSION IN A FIRST-ORDER TAYLOR SERIES ABOUT THE POINT DEFINED BY THE BEST ESTIMATE OF THE PARAMETERS. A LEAST SQUARES ANALYSIS CAN THEN BE MADE, AND THE RESULTS USED TO CORRECT THESE ESTIMATES. THIS PROCESS IS REPEATED UNTIL A SPECIFIED CONVERGENCE CRITERION IS MET.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE PROGRAM IS DESIGNED TO ACCOMMODATE UP TO 400 DATA POINTS FOR A MAXIMUM OF 10 COMPONENTS.
6. TYPICAL RUNNING TIME - A FEW SECONDS PER DECAY CURVE ARE REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM7090 VERSION SUBMITTED FEBRUARY 1968.
10. REFERENCE - PAUL C. ROGERS, FRANTIC PROGRAM FOR ANALYSIS OF EXPONENTIAL GROWTH AND DECAY CURVES, MIT LABORATORY FOR NUCLEAR SCIENCE TECHNICAL REPORT NO. 76, JUNE 1962.
11. MACHINE REQUIREMENTS - FRANTIC USES IN ITS PRESENT FORM APPROXIMATELY 12,000 MEMORY LOCATIONS.
12. PROGRAMMING LANGUAGE USED - FORTRAN II
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - FORTRAN II MONITOR.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE PROGRAM CONSISTS OF 8 SEMI-INDEPENDENT SUBPROGRAMS EACH PERFORMING A SPECIFIC SET OF OPERATIONS. THE ROUTINE CLOCK CAUSES PRINTOUT OF THE DATE AND TIME AND MAY BE DELETED.

15. NAME AND ESTABLISHMENT OF AUTHOR -

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BROOKHAVEN NATIONAL LABORATORY
UPTON, LONG ISLAND, NEW YORK

16. MATERIAL AVAILABLE -

SOURCE DECK (543 CARDS)
SAMPLE PROBLEM (30 CARDS)
REFERENCE REPORT

17. CATEGORY - P

KEYWORDS - RADIOACTIVITY, DECAY, ISOTOPES PRODUCTION, LEAST
SQUARES

REACTOR CODE ABSTRACT 325

6/68

1. NAME OR DESIGNATION OF PROGRAM - 2DB
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - 2DB IS A FLEXIBLE, TWO-DIMENSIONAL (X-Y, R-Z, R-THETA GEOMETRY) DIFFUSION CODE FOR USE IN FAST REACTOR ANALYSES. THE CODE CAN BE USED TO -
 - (A) COMPUTE FUEL BURNUP,
 - (B) PERFORM CRITICALITY SEARCHES ON TIME ABSORPTION (ALPHA), MATERIAL CONCENTRATIONS, AND REGION DIMENSIONS USING A REGULAR OR ADJOINT MODEL. CRITICALITY SEARCHES CAN BE PERFORMED DURING BURNUP TO COMPENSATE FOR FUEL DEPLETION.
4. METHOD OF SOLUTION - STANDARD SOURCE-ITERATION TECHNIQUES ARE USED. GROUP REBALANCING AND SUCCESSIVE OVERRELAXATION WITH LINE INVERSION ARE USED TO ACCELERATE CONVERGENCE. MATERIAL BURNUP IS BY REACTOR ZONE. THE BURNUP RATE IS DETERMINED BY THE ZONE AND ENERGY (GROUP) AVERAGED CROSS SECTIONS WHICH ARE RECOMPUTED AFTER EACH TIME-STEP. THE ISOTOPIC CHAINS, WHICH CAN CONTAIN ANY NUMBER OF ISOTOPES, ARE FORMED BY THE USER. THE CODE DOES NOT CONTAIN BUILT-IN OR INTERNAL CHAINS.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - SINCE VARIABLE DIMENSIONING IS EMPLOYED, NO SIMPLE BOUNDS CAN BE STATED. THE CURRENT VERSION, HOWEVER, IS NOMINALLY RESTRICTED TO 50 ENERGY GROUPS IN A 65K MEMORY.
6. TYPICAL RUNNING TIME - A KEFF CALCULATION WITH A 30 X 30 MESH AND FOUR ENERGY GROUPS REQUIRES APPROXIMATELY ONE MINUTE. A TYPICAL BURNUP CALCULATION REQUIRES ROUGHLY TWICE THE MACHINE TIME OF A KEFF CALCULATION.
7. UNUSUAL FEATURES OF THE PROGRAM - THE INPUT DATA ARE ARRANGED SO THE CODE CAN BE EASILY USED FOR KEFF AND SEARCH CALCULATIONS WITHOUT BURDENING THE USER WITH BURNUP PARAMETERS.
8. RELATED AND AUXILIARY PROGRAMS - THE INPUT DATA (E.G., CROSS SECTIONS, FLUX DUMPS, GEOMETRY, AND COMPOSITION SPECIFICATIONS) ARE COMPATIBLE WITH THE LOS ALAMOS ONE- AND TWO-DIMENSIONAL TRANSPORT CODES, DTF4 (ACC ABSTRACT 209) AND 2DF, AND THE BATTELLE PERTURBATION CODE, PERT4 (ACC ABSTRACT 304). ALL FOUR CODES USE THE SAME INPUT MODULE. THE LOGICAL FLOW OF 2DB IS PATTERNED AFTER THE 2DF CODE.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED FEBRUARY 1968.
10. REFERENCES - W. W. LITTLE, JR. AND R. W. HARDIE, 2DB, A TWO-DIMENSIONAL DIFFUSION-BURNUP CODE FOR FAST REACTOR ANALYSIS, BNWL-640, JANUARY 1968, ADDENDUM AND ERRATA, APRIL 12, 1968.

10. REFERENCES (CONTINUED)

W. W. LITTLE, JR. AND R. W. HARDIE, 2DB USERS MANUAL, BNWL-831, JULY 1968.

K. D. LATHROP, DTF-IV, A FORTRAN-IV PROGRAM FOR SOLVING THE MULTIGROUP TRANSPORT EQUATION WITH ANISOTROPIC SCATTERING, LA-3373, JULY 15, 1965.

2DF, A TWO-DIMENSIONAL TRANSPORT CODE FROM THE LOS ALAMOS SCIENTIFIC LABORATORY (UNPUBLISHED).

R. W. HARDIE AND W. W. LITTLE, JR., PERT-IV, A TWO-DIMENSIONAL PERTURBATION CODE IN FORTRAN-IV, BNWL-409, APRIL 1967.

11. MACHINE REQUIREMENTS - 65K MEMORY AND THREE PERIPHERAL STORAGE DEVICES

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (2265 CARDS)
SAMPLE PROBLEM (110 CARDS)
REFERENCE REPORTS, BNWL-640 AND BNWL-831

17. CATEGORY - D

KEYWORDS - 2-DIMENSIONAL, MULTIGROUP, REACTIVITY, CRITICALITY
SEARCHES, DEPLETION, DIFFUSION EQUATIONS, X-Y, R-Z,
R-THETA GEOMETRIES, DTF4 CODES, PERT4 CODES

REACTOR CODE ABSTRACT 326

6/68

1. NAME OR DESIGNATION OF PROGRAM - AIROS2A
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360
3. NATURE OF PHYSICAL PROBLEM SOLVED - AIROS2A SOLVES THE SPACE-INDEPENDENT REACTOR KINETICS EQUATIONS AND PROVIDES FOR THE DETERMINATION OF REACTIVITY BY SOLVING IN ADDITION THE DISCRETIZED EQUATIONS THAT REPRESENT THE SPATIAL HEAT AND MASS TRANSFER MODEL FOR SEVERAL FUEL CHANNELS. IN ADDITION, VARIATION OF THE FILM COEFFICIENT WITH FLOW IS ACCOUNTED FOR ALONG WITH THE PROVISION FOR FLOW DECAY AND AFTERGLOW HEATING. SCRAMS CAN BE INITIATED BY DELAYED SIGNALS FROM INSTRUMENTS THAT SENSE ANY QUANTITY CALCULATED, E.G., POWER, INVERSE PERIOD OR TEMPERATURE. GENERALIZED FEEDBACK EQUATIONS ARE USED TO PROVIDE FLEXIBILITY IN THE MODELS THAT REPRESENT MULTICHANNEL HEAT TRANSFER INCLUDING CONDUCTION AND CONVECTION, ENERGY, PRESSURE AND OTHER PHENOMENON SUCH AS FUEL MELTING, COOLANT BOILING AND VOIDING BURN-OUT. THE REACTIVITY EQUATION IS ALSO GENERALIZED. THE REACTIVITY FEEDBACK COEFFICIENTS CAN BE CONSTANT OR VARY AS THE SQUARE ROOT OR RECIPROCAL OF TEMPERATURE. FURTHERMORE, ANY FEEDBACK VARIABLE CAN BE USED TO INITIATE A REACTIVITY SCRAM, EACH WITH A UNIQUE DELAY TIME. AN INPUT GENERATOR COMPUTES THE CONDUCTION AND CONVECTION COEFFICIENTS FOR AN $N \times M$ NODAL, MULTICHANNEL SYSTEM USING BUILT-IN TABLES OF SPECIFIC HEAT, DENSITY, CONDUCTIVITY AND VISCOSITY FOR THE COMMON FUEL, STRUCTURE AND COOLANT MATERIALS, AND PERFORMS AN INITIAL TEMPERATURE CALCULATION. THE FILM COEFFICIENTS MAY BE SPECIFIED OR CALCULATED USING LYONS EQUATION OR THE CITTUS-BOELTER EQUATION.
4. METHOD OF SOLUTION - THE NUMERICAL TECHNIQUE USED TO INTEGRATE THE NEUTRON AND FEEDBACK DIFFERENTIAL EQUATIONS IS THAT DEVELOPED BY E. R. CHEN AS PREVIOUSLY USED IN THE AIREK CODES. AN IMPROVED INTERVAL SWITCHING TECHNIQUE ALLOWS RAPID CALCULATIONS WITH PRE-DETERMINED ACCURACY.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
15 DELAYED NEUTRON PRECURSOR GROUPS
400 FEEDBACK VARIABLES
90 FEEDBACK VARIABLES PRINTED OUT
ANY NUMBER OF CHANNELS AND NODES PER CHANNEL WITHIN THE LIMITATION ABOVE ARE ALLOWED.
6. TYPICAL RUNNING TIME - 1 TO 2 MINUTES ARE REQUIRED FOR A PROBLEM WITH 6 DELAYED NEUTRON PRECURSOR GROUPS AND 50 FEEDBACK VARIABLES (INCLUDING CRT).
7. UNUSUAL FEATURES OF THE PROGRAM -
 - (A) SPECIAL PROVISION MADE FOR REACTOR STARTUP PROBLEMS RESULTING IN A LARGE REDUCTION IN RUNNING TIME.
 - (B) MANY OF THE REQUIRED INPUT DATA ARE PRE-SET BUT CAN BE CHANGED IF DESIRED.

7. UNUSUAL FEATURES OF THE PROGRAM (CONTINUED)
 - (C) ADDRESSABLE INPUT DATA ARE USED SO THAT ON MULTIPLE CASES, ONLY CHANGES NEED BE SPECIFIED.
 - (D) A RESTART FEATURE IS PROVIDED WHEREIN RESTART CARDS ARE PUNCHED UPON ABNORMAL PROBLEM TERMINATION AND/OR ON AN INPUT OPTION.
 - (E) EXTENSIVE PRINTED AND GRAPHICAL DISPLAYS ARE PROVIDED AS FOLLOWS - POWER, INVERSE PERIOD, REACTIVITY AND ANY 90 FEEDBACK VARIABLES. PRINTING AND DISPLAY OF FEEDBACK VARIABLES IS UNDER THE USERS CONTROL AND THE LATTER CAN BE GROUPED ON CRT FRAMES AS DESIRED.
 - (F) CONDUCTION AND CONVECTION COEFFICIENTS MAY BE INPUT AND/OR CALCULATED BY MEANS OF AN INPUT GENERATOR WHICH IN ADDITION PERFORMS AN INITIAL TEMPERATURE CALCULATION.
 - (G) PHENOMENOLOGICAL MODELS FOR CHANGE OF PHASE ARE INCORPORATED.
8. RELATED AND AUXILIARY PROGRAMS - AIROS2A REPLACES THE 360 VERSION OF AIROS (ACC ABSTRACT 163).
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM360 VERSION SUBMITTED FEBRUARY 1968.
10. REFERENCES - R. A. BLAINE AND R. F. BERLAND, SIMULATION OF REACTOR DYNAMICS, VOLUME I - A DESCRIPTION OF AIROS IIA, NAA-SR-12452, SEPTEMBER 1967.
R. A. BLAINE, AI ENVIRONMENT REPORT, MAY 1967.
11. MACHINE REQUIREMENTS - 256K BYTE IBM360 AND AN SC-4020 GRAPHICAL DISPLAY DEVICE
12. PROGRAMMING LANGUAGES USED - FORTRAN IV (95 PER CENT)
360 ASSEMBLY LANGUAGE (5 PER CENT)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - IF AN SC-4020 GRAPHICAL DISPLAY DEVICE IS NOT AVAILABLE, THE AICRT 3 DISPLAY ROUTINE CAN BE REWRITTEN, OR ALL ROUTINES BUT TAPOUT AND PRINT CAN BE DELETED FROM THE LAST LINK. THE NAA SC-4020 (OS) SUBROUTINE PACKAGE IS AVAILABLE THROUGH -
CAIDE LIBRARIAN
C/O STROMBERG-CARLSON
P. O. BOX 2449
SAN DIEGO, CALIFORNIA 92112

REACTOR CODE ABSTRACT 326

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6/68

15. NAME AND ESTABLISHMENT OF AUTHOR -

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P. O. BOX 309
CANGA PARK, CALIFORNIA 91304

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL

SOURCE DECK (4204 CARDS)
SAMPLE PROBLEM (55 CARDS)
REFERENCE REPORTS

17. CATEGORY - E

KEYWORDS - SPACE-INDEPENDENT KINETICS, REACTIVITY, FEEDBACK EQUATIONS, GRAPHS, AIREK CODES

1. NAME OR DESIGNATION OF PROGRAM - DAFT1
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - DAFT1 IS A PROGRAM FOR WEIGHTED LEAST SQUARES FITTING OF 0.0253 EV NEUTRON DATA FOR FISSIONABLE NUCLIDES. THE PROGRAM ALSO CARRIES OUT COMPUTATIONS RELEVANT TO DISCERNING OVERALL GOODNESS OF FIT, PARTICULARLY DEVIANT DATA, AND DATA WHOSE IMPROVEMENT WOULD LEAD TO LARGE REDUCTIONS IN ERROR OF EACH FITTED PARAMETER.
4. METHOD OF SOLUTION - NORMAL EQUATIONS ARE SOLVED BY GAUSS-NEWTON ITERATION USING THE PIVOTAL METHOD FOR MATRIX INVERSION. THE CHOICE OF INPUT AND FITTED PARAMETERS FOLLOWS REFERENCE 1.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - DATA ARE FITTED FOR A MAXIMUM OF 3 FISSIONABLE NUCLIDES.
6. TYPICAL RUNNING TIME - 30 SECONDS ARE REQUIRED FOR FOUR ITERATIONS.
7. UNUSUAL FEATURES OF THE PROGRAM - COMPUTATIONS ARE CARRIED OUT TO AID IN DIAGNOSING PARTICULARLY DEVIANT DATA AND PARTICULARLY SIGNIFICANT DATA.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC6600 VERSION SUBMITTED MARCH 1968.
10. REFERENCES - C. H. WESTCOTT, ET AL., A SURVEY OF VALUES OF THE 2200 M/S CONSTANTS FOR FOUR FISSIONABLE NUCLIDES, ATOMIC ENERGY REVIEW, VOL. 3, NO. 2, P. 1, 1965.
D. R. HARRIS, DAFT1 - A FORTRAN PROGRAM FOR LEAST SQUARES FITTING OF 0.0253 EV NEUTRON DATA FOR FISSIONABLE NUCLIDES, WAPD-TM-761, FEBRUARY 1968.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
11. MACHINE REQUIREMENTS - 32000 (BASE 8) CENTRAL MEMORY LOCATIONS
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -

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WESTINGHOUSE ELECTRIC CORPORATION
BETTIS ATOMIC POWER LABORATORY
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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION

MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (360 CARDS)
SAMPLE PROBLEM (17 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS, WAPD-TM-761 AND WAPD-TM-668

17. CATEGORY - D

KEYWORDS - LEAST SQUARES, CROSS SECTIONS, URANIUM, PLUTONIUM

1. NAME OR DESIGNATION OF PROGRAM - NURLOC-1.0
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6400
3. NATURE OF PHYSICAL PROBLEM SOLVED - NURLOC-1.0 PERFORMS CORE/PRESSURE VESSEL THERMAL ANALYSIS FOR A NUCLEAR REACTOR LOSS-OF-COOLANT ACCIDENT.
4. METHOD OF SOLUTION - EXPLICIT FINITE DIFFERENCE APPROXIMATION IS USED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - TWO-DIMENSIONAL ANALYSES, EITHER SINGLE PINS OR CYLINDRICAL SHELLS
6. TYPICAL RUNNING TIME - A TYPICAL PROBLEM WITH 1000 TIME-STEPS REQUIRES ABOUT 20 MINUTES ON THE CDC6400.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - ECCSA-I (ACC ABSTRACT 330), A BLOWDOWN AND EMERGENCY-CORE-COOLING HEAT TRANSFER PROGRAM
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC6400 VERSION SUBMITTED APRIL 1968.
10. REFERENCE - C. T. WALTERS AND J. M. GENCO, NURLOC-1.0 A DIGITAL COMPUTER PROGRAM FOR THERMAL ANALYSIS OF A NUCLEAR-REACTOR LOSS-OF-COOLANT ACCIDENT, BMI-1807, JULY 6, 1967.
11. MACHINE REQUIREMENTS - 65K OR 32K MEMORY WITH OVERLAY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (6400-3847 CARDS)
SAMPLE PROBLEM (6400-271 CARDS)
REFERENCE REPORT

REACTOR CODE ABSTRACT 328

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6/68

17. CATEGORY - G
KEYWORDS - 2-DIMENSIONAL, SHELLS, WATER REACTORS, ACCIDENTS,
COOLANTS, THERMAL, TEMPERATURE DISTRIBUTION

1. NAME OR DESIGNATION OF PROGRAM - M0457(PIPE)
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - PIPE PERFORMS AN ELASTIC STRESS ANALYSIS OF A 3-DIMENSIONAL PIPING STRUCTURE WITH THERMAL STRESSES, REDUNDANT LOOPS, AND CONCENTRATED LOADS.
4. METHOD OF SOLUTION - LINEAR ELASTIC THEORY IS USED. EQUATIONS ARE SET UP IN ACCORDANCE WITH CTAC AND TRAIL THEORY.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
 - 100 POSITIONS TO LOCATE FORCES AND ELEMENTS
 - 20 FORCE-TRAJLS
 - 25 APPLIED LOADS
 - 50 TOTAL LOADS
 - 20 JOINT-CONDITION LOADS
 - 20 SETS OF MATERIAL PROPERTIES
 - 20 SETS OF STIFFNESS DATA
 - 99 ELASTIC ELEMENTS
6. TYPICAL RUNNING TIME - ONE MINUTE IS REQUIRED.
7. UNUSUAL FEATURES OF THE PROGRAM - TRAIL THEORY ALLOWS THE USER TO INPUT GEOMETRIC DATA FROM WHICH THE PROGRAM CALCULATES CONNECTION MATRICES FOR THE CTAC EQUATIONS.
8. RELATED AND AUXILIARY PROGRAMS - PIPE IS A SIMPLIFICATION OF THE MORE BASIC PROGRAM M0014. PIPE APPLIES MORE DIRECTLY TO PIPING SYSTEMS.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC6600 VERSION SUBMITTED MARCH 1968.
10. REFERENCES - C. M. FRIEDRICH, PIPE - A COMPUTER PROGRAM FOR THE ELASTIC STRESS ANALYSIS OF A PIPING STRUCTURE WITH THERMAL STRAINS AND EXTERNAL LOADS, WAPD-TM-752, MARCH 1968.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
11. MACHINE REQUIREMENTS - 100,000 (BASE 8) CENTRAL MEMORY AND 3 TAPES (INPUT, OUTPUT, STORAGE)
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

REACTOR CODE ABSTRACT 329

-2-

6/68

15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION

MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (1632 CARDS)
SAMPLE PROBLEM (47 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS

17. CATEGORY - I

KEYWORDS - ELASTICITY, THERMAL STRESSES, PIPES, 3-DIMENSIONAL,
PRESSURE

1. NAME OR DESIGNATION OF PROGRAM - ECCSA1
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6400
3. NATURE OF PHYSICAL PROBLEM SOLVED - ECCSA1 PREDICTS THE THERMAL AND HYDRAULIC PERFORMANCE OF REACTOR CORE CHANNELS DURING A LOSS-OF-COOLANT ACCIDENT AND SUBSEQUENT EMERGENCY CORE COOLING INJECTION.
4. METHOD OF SOLUTION - EXPLICIT SOLUTION OF THE TIME-DEPENDENT EQUATIONS IS USED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - EMERGENCY CORE COOLANT MAY ENTER EITHER AT THE TOP OR AT THE BOTTOM OF THE CHANNEL BUT NOT AT BOTH ENDS SIMULTANEOUSLY.
6. TYPICAL RUNNING TIME - APPROXIMATELY 1000 TIME-STEPS CAN BE DONE IN 1 MINUTE ON THE CDC6400.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - NURLOC-1.0 (ACC ABSTRACT 328)
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC6400 VERSION SUBMITTED APRIL 1968.
10. REFERENCES - J. IYER, ECCSA-I A DIGITAL COMPUTER PROGRAM FOR THERMAL AND HYDRAULIC ANALYSIS OF CORE CHANNELS IN THE EVENT OF A NUCLEAR-REACTOR LOSS-OF-COOLANT ACCIDENT, BMI-1832, APRIL 1, 1968.
CRAIG T. WALTERS AND JOSEPH M. GENCO, NURLOC-1.0 A DIGITAL COMPUTER PROGRAM FOR THERMAL ANALYSIS OF A NUCLEAR-REACTOR LOSS-OF-COOLANT ACCIDENT, BMI-1807, JULY 6, 1967.
11. MACHINE REQUIREMENTS - 64K MEMORY WITH 2 TAPE UNITS
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCCPE 2.0.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
J. S. IYER
BATTELLE MEMORIAL INSTITUTE
COLUMBUS LABORATORIES
505 KING AVENUE
COLUMBUS, OHIO 43201

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (6400-3789 CARDS)
SAMPLE PROBLEM (6400-134 CARDS)
REFERENCE REPORTS
17. CATEGORY - G
KEYWORDS - ACCIDENTS, THERMODYNAMICS PERFORMANCE, COOLANTS, TEMPERATURE DISTRIBUTION, HEAT TRANSFER, BLOWDOWN, PRESSURE, BWR REACTORS, PWR REACTORS

1. NAME OR DESIGNATION OF PROGRAM - M0219(FLOT1)
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600
3. NATURE OF PHYSICAL PROBLEM SOLVED - FLOT1 WILL PREDICT THE STEADY-STATE FLOW AND THE FLOW TRANSIENT DUE TO THE SUBSEQUENT LOSS OF POWER TO ALL PUMPS AND TERMINATE THE TRANSIENT AT A SPECIFIED TIME OR IT WILL PREDICT THE FLOW TRANSIENT IN WHICH ONLY SOME OF THE PUMPS ARE LOST. THIS LATTER TRANSIENT MAY BE TERMINATED BY A MAXIMUM TRANSIENT TIME OR BY CHECK VALVE CLOSURES IN ALL LOOPS IN WHICH PUMPING POWER IS LOST. IN THE LATTER EVENT, THE PROGRAM WILL PREDICT THE SUBSEQUENT STEADY-STATE FLOW DISTRIBUTION.
4. METHOD OF SOLUTION - THE SIMULTANEOUS DIFFERENTIAL EQUATIONS OF FLOW ARE SOLVED BY A MATRIX AND, IN CONJUNCTION WITH DYNAMIC EQUATIONS OF THE PUMP, ARE ITERATED TO USER SPECIFIED CONVERGENCE CRITERION.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE MATHEMATICAL MODEL, UPON WHICH THE PROGRAM IS BASED, IS LIMITED TO FOUR LOOPS WITH A MAXIMUM OF THREE PUMPS PER LOOP IN PARALLEL.
6. TYPICAL RUNNING TIME - THE RUNNING TIME FOR A THREE LOOP INITIAL OPERATION TO A COMPLETE LOSS OF FLOW REQUIRES 49 SECONDS WITH OUTPUT CONSISTING OF INITIAL STEADY-STATE FLOW AND PRESSURE-DROP DISTRIBUTION AND A PRINTOUT EVERY 50 MILLISECONDS FOR 10 SECONDS.
7. UNUSUAL FEATURES OF THE PROGRAM - EMPIRICAL RELATIONS FOR THE TRANSFER OF ENERGY FROM THE PUMP IMPELLER TO THE FLUID OVER THE ENTIRE RANGE OF OPERATION HAVE BEEN INCLUDED IN THE PROGRAM AND THE PROGRAM RESULTS SHOW GOOD CORRELATION WITH TEST DATA. THE PROGRAM RESULTS ARE USED IN CORE PERFORMANCE ANALYSIS AND CHECK VALVE PRESSURE SURGE CALCULATIONS.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC6600 VERSION SUBMITTED MAY 1968.
10. REFERENCES - G. M. FULS, FLOT-1 FLOW TRANSIENT ANALYSIS OF A PRESSURIZED WATER REACTOR DURING FLOW COASTDOWN, WAPD-TM-428, APRIL 1968.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
11. MACHINE REQUIREMENTS - THE PROGRAM IS DESIGNED FOR CARD INPUT WITH BOTH PRINTED AND PLOTTED OUTPUT. THE NUMBER OF CORE LOCATIONS REQUIRED IS 55000 OCTAL.
12. PROGRAMMING LANGUAGE USED - FORTRAN IV

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13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
SCOPE 2.C.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
G. M. FULS
WESTINGHOUSE ELECTRIC CORPORATION
BETTIS ATOMIC POWER LABORATORY
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122
16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (2126 CARDS)
SAMPLE PROBLEM (36 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,125 CARDS)
REFERENCE REPORTS
17. CATEGORY - I
KEYWORDS - FLUID FLOW, PUMPS, STEAM GENERATORS, POWER FAILURES,
PWR REACTORS

1. NAME OR DESIGNATION OF PROGRAM - SAFE-3D
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - SAFE-3D IS A FINITE ELEMENT PROGRAM FOR THE THREE-DIMENSIONAL ELASTIC ANALYSIS OF HETEROGENEOUS COMPOSITE STRUCTURES. THE PROGRAM USES THE FOLLOWING TYPES OF FINITE ELEMENTS - (1) TETRAHEDRAL ELEMENTS TO REPRESENT THE CONTINUUM, (2) TRIANGULAR PLANE STRESS MEMBRANE ELEMENTS TO REPRESENT INNER LINER OR OUTER CASE, AND (3) UNIAXIAL TENSION-COMPRESSION ELEMENTS TO REPRESENT INTERNAL REINFORCEMENT. THE STRUCTURE CAN BE OF ARBITRARY GEOMETRY AND HAVE ANY DISTRIBUTION OF MATERIAL PROPERTIES, TEMPERATURES, SURFACE LOADINGS, AND BOUNDARY CONDITIONS.
4. METHOD OF SOLUTION - THE FINITE ELEMENT VARIATIONAL METHOD IS USED. EQUILIBRIUM EQUATIONS ARE SOLVED BY THE ALTERNATING COMPONENT ITERATIVE METHOD.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
5000 NODES
16000 ELEMENTS
THE PROGRAM CANNOT BE APPLIED TO INCOMPRESSIBLE SOLIDS AND IS NOT RECOMMENDED FOR POISSONS RATIO IN THE RANGE OF ν BETWEEN 0.495 AND 0.5.
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - SAFE-PCRS (ACC ABSTRACT 250), SAFE-AXISYM (ACC ABSTRACT 251), SAFE-PLANE (ACC ABSTRACT 252)
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED APRIL 1968.
10. REFERENCES - D. C. CORNELL, K. B. JADHAV, AND Y. R. RASHID, SAFE-3D, A COMPUTER PROGRAM FOR THE THREE-DIMENSIONAL STRESS ANALYSIS OF COMPOSITE STRUCTURES - A USERS MANUAL, GA-7855, SEPTEMBER 1, 1967.
Y. R. RASHID, THREE-DIMENSIONAL ANALYSIS OF ELASTIC SOLIDS, GA-8419, FEBRUARY 1968.
11. MACHINE REQUIREMENTS - 52K FAST MEMORY STORAGE AND 8,000,000 AUXILIARY STORAGE
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II, GAX33A.

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14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
D. C. CORNELL AND K. JADHAV
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (5151 CARDS)
SAMPLE PROBLEM (207 CARDS)
REFERENCE REPORT, GA-7855
17. CATEGORY - I
KEYWORDS - 3-DIMENSIONAL, STRUCTURAL ANALYSIS, STRESSES,
FINITE ELEMENT

1. NAME OR DESIGNATION OF PROGRAM - TOAD
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108
3. NATURE OF PHYSICAL PROBLEM SOLVED - TOAD IS USED TO PROCESS AND ANALYZE GAMMA RAY SPECTRA.
4. METHOD OF SOLUTION - SPECTRAL DATA IS READ FROM MAGNETIC (BCD) TAPE, PAPER TAPE, OR CARDS, AND SMOOTHED. PEAKS ARE LOCATED AND INTEGRATED. PEAK ENERGIES ARE DETERMINED. ADDITIONAL CALCULATIONS ARE PERFORMED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - TOAD WILL HANDLE UP TO 100 SPECTRA AT A TIME FROM ANALYZERS WITH 4096 CHANNELS OR FEWER.
6. TYPICAL RUNNING TIME - 30 SECONDS TO 5 MINUTES ARE REQUIRED, DEPENDING UPON THE NUMBER AND COMPLEXITY OF SPECTRA PROCESSED.
7. UNUSUAL FEATURES OF THE PROGRAM - TOAD IS TOLERANT OF ANALYZER GAIN AND BASE LINE SHIFTS AND CHANGES IN SAMPLE GEOMETRY AND DOES NOT REQUIRE A LIBRARY OF STANDARD SPECTRA.
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
UNIVAC1108 VERSION SUBMITTED APRIL 1968.
10. REFERENCES - J. DRAKE, L. L. STEWART, AND D. D. BUSCH, TOAD, A COMPUTER CODE FOR PROCESSING GAMMA-RAY SPECTRA, GAMD-8266, APRIL 3, 1968.
H. P. YULE, DATA CONVOLUTION AND PEAK LOCATION, PEAK AREA, AND PEAK ENERGY MEASUREMENTS IN SCINTILLATION COUNTING, ANALYTICAL CHEMISTRY, VOL. 38, NO. 1, P. 103, JANUARY 1, 1966.
11. MACHINE REQUIREMENTS - 32K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II, GAX33.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
D. D. BUSCH
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (1835 CARDS)
SAMPLE PROBLEM (65 CARDS)
4096 CHANNEL ANALYZER DATA FOR SAMPLE PROBLEM (FILE 2-16 BCD
RECORDS OF LENGTH 1542 BCD, FILE 3-16 BCD RECORDS OF
LENGTH 1542 BCD)
REFERENCE REPORT, GAMC-8266
17. CATEGORY - 0
KEYWORDS - GAMMA RADIATION SPECTRA, DATA PROCESSING

1. NAME OR DESIGNATION OF PROGRAM - PEGGY
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC1604
3. NATURE OF PHYSICAL PROBLEM SOLVED - PEGGY IS A LEAST SQUARES SEARCH PROGRAM WHICH ANALYZES, IN TERMS OF PHASE SHIFTS, THE ELASTIC SCATTERING OF SPIN ZERO AND SPIN ONE-HALF PARTICLES BY SPIN ZERO NUCLEI. REAL OR COMPLEX PHASE SHIFTS MAY BE USED WITH OR WITHOUT SPIN-ORBIT COUPLING. DIFFERENTIAL CROSS SECTION AND POLARIZATION ANGULAR DISTRIBUTIONS MAY BE ANALYZED EITHER SEPARATELY OR SIMULTANEOUSLY.
4. METHOD OF SOLUTION - MATRIX INVERSION UTILIZING AN ORNL ADAPTATION OF THE CC-OP MATALG SUBROUTINE IS USED.
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF - 70 DATA POINTS
6. TYPICAL RUNNING TIME - RUNNING TIME PER CASE IS ABOUT 20 SECONDS FOR ABOUT TWENTY OR TWENTY-FIVE DATA POINTS, EMPLOYING THREE SEARCH MODES, VARYING AN AVERAGE OF TWO PARAMETERS PER MODE, AND USING REAL PHASE SHIFTS WITH SPIN-ORBIT COUPLING.
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC1604 VERSION SUBMITTED APRIL 1968.
10. REFERENCE - L. W. OWEN, PHASE SHIFT ANALYSIS OF ELASTIC SCATTERING, THE CODE PEGGY, ORNL-TM-1464, MARCH 16, 1966, AMENDED TO INCLUDE INPUT CHANGES.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN 63
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
L. W. OWEN
OAK RIDGE NATIONAL LABORATORY
P. O. BOX X
OAK RIDGE, TENNESSEE 37831

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16. MATERIAL AVAILABLE -
SOURCE DECK (1326 CARDS)
SAMPLE PROBLEM (48 CARDS)
REFERENCE REPORT
17. CATEGORY - A
KEYWORDS - LEAST SQUARES, SEARCHES, ELASTIC SCATTERING, ANGULAR
DISTRIBUTION, CROSS SECTIONS

1. NAME OR DESIGNATION OF PROGRAM - RAMES
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC1604
3. NATURE OF PHYSICAL PROBLEM SOLVED - RAMES COMPUTES BOTH LOCAL AND NONLOCAL RADIAL INTEGRALS OF A VARIETY OF RADIAL OPERATORS USING SINGLE-PARTICLE WAVE FUNCTIONS WHICH ARE EIGENSTATES OF MOTION IN A WOODS-SAXON POTENTIAL WELL. THE OPERATORS CURRENTLY AVAILABLE ARE R^{*N} , $N=0,1,2,3,4,5$ AND THE DERIVATIVE WITH RESPECT TO X OF $1/(\exp(X)+1)$ WHERE $X=(R-R(0))/A(0)$.
4. METHOD OF SOLUTION -
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
6. TYPICAL RUNNING TIME -
7. UNUSUAL FEATURES OF THE PROGRAM -
8. RELATED AND AUXILIARY PROGRAMS - RAMES IS A REVISION OF THE PROGRAM, OVERLAP.
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
CDC1604 VERSION SUBMITTED APRIL 1968.
10. REFERENCE - L. W. OWEN, RADIAL INTEGRALS USING REALISTIC SINGLE-PARTICLE WAVE FUNCTIONS AND THE CODE OVERLAP, ORNL-TM-958, SEPTEMBER 30, 1964.
11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN 63
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
L. W. OWEN
OAK RIDGE NATIONAL LABORATORY
P. O. BOX X
OAK RIDGE, TENNESSEE 37831
16. MATERIAL AVAILABLE -
SOURCE DECK (1169 CARDS)
SAMPLE PROBLEM (15 CARDS)
REFERENCE REPORT
17. CATEGORY - A
KEYWORDS - OVERLAP CODES, CRCSS SECTIONS, OPTICAL MODEL

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1. NAME OR DESIGNATION OF PROGRAM - PDQ5
2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360
3. NATURE OF PHYSICAL PROBLEM SOLVED - THE FEW-GROUP TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED. UP TO FIVE GROUPS MAY BE USED WITH SCATTERING ALLOWED BETWEEN ADJACENT GROUPS. IN ADDITION, DEPLETION PROBLEMS MAY BE SOLVED WITH PDQ5.
4. METHOD OF SOLUTION -
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE NUMBER OF COMPOSITIONS (WHICH IS ACTUALLY THE LARGEST COMPOSITION NUMBER PRESENT IN THE INPUT) MUST BE BETWEEN 1 AND 100. IT IS NOT NECESSARY THAT EVERY COMPOSITION APPEAR IN THE MESH. THE NUMBER OF EDIT REGIONS (LARGEST EDIT NUMBER PRESENT IN THE INPUT) MUST BE BETWEEN 0 AND 500. IF 0, INTEGRATION EDITING IS DONE BY COMPOSITION RATHER THAN BY EDIT REGION, IT IS NOT NECESSARY THAT EVERY EDIT REGION APPEAR IN THE MESH. THE ROW AND COLUMN BOUNDARIES MAY NOT CHANGE BETWEEN CASES OF A JOB. THE NUMBER OF SOLUTION COLUMNS IS GIVEN BY (COLUMN BOUNDARY + LEFT BOUNDARY CONDITION + RIGHT BOUNDARY CONDITION - 1) AND THE NUMBER OF SOLUTION ROWS BY (ROW BOUNDARY + TOP BOUNDARY CONDITION + BOTTOM BOUNDARY CONDITION - 1) WHERE THE BOUNDARY CONDITIONS ARE -1, 0 AND +1 FOR ROTATION, ZERO FLUX, AND ZERO DERIVATIVE, RESPECTIVELY. THE NUMBER OF SOLUTION COLUMNS MUST BE BETWEEN 3 AND 335. THE NUMBER OF SOLUTION ROWS MUST BE EXACTLY 2 IN ONE-DIMENSIONAL PROBLEMS AND BE GREATER THAN OR EQUAL TO 3 AND APPROXIMATELY 440 IN ALL OTHER PROBLEMS. IN R-Z GEOMETRY, THE TOP BOUNDARY CONDITION MAY NOT BE IN ROTATION AND THE LEFT BOUNDARY CONDITION MUST BE ZERO DERIVATIVE. THE LEFT AND RIGHT BOUNDARY CONDITIONS MUST BE THE SAME IF THE TOP BOUNDARY CONDITION IS ROTATION. THE TOP AND BOUNDARY CONDITIONS MUST BE ZERO DERIVATIVE IN ONE-DIMENSIONAL PROBLEMS. THE TOTAL NUMBER OF GROUPS MUST BE BETWEEN 1 AND 5 AND MAY NOT BE CHANGED BETWEEN CASES OF A JOB. THE NUMBER OF THERMAL GROUPS IS RESTRICTED TO 1.
6. TYPICAL RUNNING TIME - ON AN IBM360, MODEL 65, THE RANGE MAY BE FROM ONE MINUTE TO TWO HOURS WITH TYPICAL PROBLEMS IN THE 5 TO 30 MINUTE RANGE.
7. UNUSUAL FEATURES OF THE PROGRAM -
 - (A) SENSE SWITCH OPERATION VIA TYPEWRITER
 - (B) ADAPTABILITY TO WIDE CONFIGURATION OF EQUIPMENTS
 - (C) FULL DOUBLE-PRECISION REAL VARIABLES
8. RELATED AND AUXILIARY PROGRAMS -
9. STATUS - ABSTRACT FIRST DISTRIBUTED JUNE 1968.
IBM360 VERSION SUBMITTED APRIL 1968.

10. REFERENCES - THE PDQ-5 PROGRAM FOR THE SOLUTION OF THE TWO-DIMENSIONAL NEUTRON DIFFUSION-DEPLETION PROBLEM USING IBM SYSTEM 360, IBM REFERENCE MANUAL, JANUARY 1968.
PDQ-5 USER'S OPERATING NOTES, IBM NOTES.
11. MACHINE REQUIREMENTS - THE MINIMUM REQUIREMENTS FOR A MODEL 40 CPU ARE 230K BYTES OF CORE FOR PROGRAM RESIDENCE, 50 CYLINDERS OF A 2311 OR 2314 DISK PACK FOR LOAD MODULE RESIDENCE, THE ABILITY TO HANDLE 13 DATA SETS (ON 2400 TAPE DRIVES OR 2311 OR 2314 DISK DRIVES) IN ADDITION TO STANDARD INPUT AND STANDARD OUTPUT. A TYPICAL SYSTEM 360 CONFIGURATION IS A MODEL 65 WITH AT LEAST 256K BYTES OF MEMORY, 3 MODEL 2311 DISK PACKS (2 BEING SYSTEM RESIDENCE PACKS AND THE THIRD CONTAINING THE PDQ5 LOAD MODULE AND ONE DATA SET), 13 MODEL 2402 9-TRACK TAPE DRIVES FOR 12 OF THE PROBLEM DATA SETS AND SYSOUT, AND A MODEL 2540 CARD READER.
12. PROGRAMMING LANGUAGES USED - FORTRAN IV (H) AND B.A.L. (IBM BASIC ASSEMBLY LANGUAGE)
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360, RELEASE 12, OR LATER.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
S. G. REED
DATA PROCESSING DIVISION
INTERNATIONAL BUSINESS MACHINES CORPORATION
112 EAST POST ROAD
WHITE PLAINS, NEW YORK 10601
16. MATERIAL AVAILABLE -
SOURCE DECK
SAMPLE PROBLEM
REFERENCE MANUAL AND NOTES
17. CATEGORY - D
KEYWORDS - DEPLETION, 2-DIMENSIONAL, FEW-GROUP, DIFFUSION EQUATIONS, REACTIVITY, FLUX DISTRIBUTION

VI. LIBRARY PROGRAMS BY CLASSIFICATION

A. CROSS SECTION AND RESONANCE INTEGRAL CALCULATIONS

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14 HAFEVER HAFEVER CALCULATES THE ENERGY EXCHANGE INELASTIC SCATTERING CROSS SECTION ACCORDING TO THE HAUSER-FESHBACH THEORY MODIFIED TO INCLUDE THE EFFECT OF SPIN-ORBIT COUPLING. THE CROSS SECTION IS INTEGRATED OVER ANGLE.

41 ZUT ZUT COMPUTES RESONANCE INTEGRALS FROM RESONANCE PARAMETERS FOR A WIDE VARIETY OF TEMPERATURES, COMPOSITIONS, AND GEOMETRIES FOR THE RESOLVED RESONANCES.

42 TUZ TUZ COMPUTES RESONANCE INTEGRALS FOR A WIDE VARIETY OF TEMPERATURES, COMPOSITIONS, AND GEOMETRIES FOR THE UNRESOLVED RESONANCES.

56 SUMMIT THE PROGRAM EVALUATES THE DIFFERENTIAL ENERGY-TRANSFER CROSS SECTION FOR SCATTERING BY A CRYSTALLINE MODERATOR, UTILIZING THE SO-CALLED PHONON EXPANSION. THE SCATTERING KERNEL FOR A 1-PHONON CHANGE IN ENERGY IS ADDED TO THAT FOR A 2-PHONON ENERGY EXCHANGE, AND SO ON. THIS PROGRAM HAS BEEN USED TO DETERMINE SCATTERING MATRICES FOR BERYLLIUM, GRAPHITE, AND OXYGEN.

$$\text{SIGMA}(E(0) \text{ TO } E)/\text{SIGMA}(0) = (((M+1)/M)**2) * \text{SQRT}(E/E(0)) * 1/2$$

 THE INTEGRAL FROM -1 TO 1 OF $\text{SIGMA}(E(0) \text{ TO } E, \cos(\theta))$
 $D(\cos(\theta))$

WHERE $E(0)$ AND E ARE THE INITIAL AND FINAL ENERGIES, θ IS THE ANGLE OF SCATTERING, $\text{SIGMA}(0)$ THE FREE-ATOM CROSS SECTION, AND M THE RATIO OF THE MASS OF THE SCATTERING NUCLEUS TO THAT OF THE NEUTRON.

89 ARES2 ARES2 IS USED TO CALCULATE EFFECTIVE RESONANCE INTEGRALS AND MULTIGROUP CROSS SECTIONS FOR LUMPS AND MIXTURES USING RESONANCE PARAMETERS. IT COMBINES IN A SINGLE CODE, THE RESOLVED, UNRESOLVED, AND $1/V$ PARTS OF THE CALCULATION WHICH WERE PREVIOUSLY IN SEPARATE CODES. THE MULTIGROUP CROSS SECTIONS ARE PRINTED IN A FORM FOR USE IN MULTIGROUP REACTOR CALCULATIONS. FOR EXAMPLE, THEY CAN BE INSERTED INTO THE AIM6 (ACC ABSTRACT 29) OR FAIM (ACC ABSTRACT 120) LIBRARIES. THE ENERGY BREAKPOINTS ARE INPUT DATA AND A RESONANCE PARAMETER LIBRARY IS INCLUDED.

171 NEARREX NEARREX COMPUTES NEUTRON-INDUCED, AVERAGE FLUCTUATION (OR COMPOUND NUCLEUS) CROSS SECTIONS. PROVISION IS MADE FOR THE COMPUTATION OF COMPOUND ELASTIC AND INELASTIC NEUTRON CROSS SECTIONS, RADIATIVE CAPTURE AND FISSION CROSS SECTIONS, AS WELL AS OTHER PROCESSES, SUCH AS PROTON EMISSION. IT CAN ALSO BE USED TO COMPUTE PROTON-INDUCED AVERAGE CROSS SECTIONS.

215 CHAD CHAD IS DESIGNED TO FACILITATE ANALYSIS AND HANDLING OF DIFFERENTIAL NEUTRON SCATTERING DATA. IT PRODUCES LEGENDRE SCATTERING COEFFICIENTS FROM ANGULAR DISTRIBUTION DATA TABULATED IN MANY DIFFERENT FORMATS. IT CAN TRANSFORM ANGULAR DATA INTO LEGENDRE SCATTERING COEFFICIENTS IN EITHER THE LABORATORY OR THE CENTER-OF-MASS FRAME OF REFERENCE. IT CALCULATES THE AVERAGE COSINE OF THE SCATTERING ANGLE IN THE LABORATORY SYSTEM AND THE AVERAGE LOGARITHMIC ENERGY DECREMENT PER ELASTIC COLLISION.

216 FASDOP FASDOP EVALUATES POINTWISE, TEMPERATURE-DEPENDENT CROSS SECTIONS FROM SINGLE-LEVEL BREIT-WIGNER RESONANCE PARAMETERS.

217 LEGCOEF3 CALCULATES LEGENDRE EXPANSION COEFFICIENTS FOR THE ANGULAR DISTRIBUTION OF ELASTICALLY SCATTERED NEUTRONS.

219 GAROL GAROL COMPUTES EFFECTIVE GROUP CROSS SECTIONS FOR THE RESOLVED RESONANCES OF A MIXTURE OF ISOTOPES IN A TWO-REGION CELL. BASIC CROSS SECTIONS INCLUDE TEMPERATURE DEPENDENCE. THE PROGRAM ALLOWS A CHOICE OF GEOMETRIES AND CAN ACCEPT AN ARBITRARY TABLE OF ESCAPE PROBABILITIES. A DANCUFF CORRECTION MAY BE USED TO ACCOUNT FOR SHADOWING EFFECTS IN A TIGHT LATTICE, AND CROSS SECTIONS MAY BE $1/v$, CONSTANT, COMPUTED FROM BREIT-WIGNER RESONANCE PARAMETERS, OR GIVEN IN TABULAR FORM. THE MESH MAY BE CHOSEN AT EQUAL ENERGY OR LETHARGY INTERVALS, PROPORTIONAL TO THE NEUTRON VELOCITY, OR AS AN ARBITRARY TABLE OF VALUES.

238 EXT/XO GIVEN A SET OF BREIT-WIGNER RESONANCE PARAMETERS (ZERO TEMPERATURE) EXT CALCULATES THE EFFECTIVE DOPPLER-BROADENED CROSS SECTIONS FOR ANY TEMPERATURE. THE EFFECTIVE CROSS SECTION WHEN MULTIPLIED BY THE TARGET DENSITY AND THE NEUTRON VELOCITY (LAB) GIVES THE REACTION RATE PER INCIDENT NEUTRON. A MAXWELLIAN VELOCITY DISTRIBUTION IS ASSUMED FOR THE TARGET NUCLEI. THE ANALYSIS IS CARRIED OUT FOR SUFFICIENTLY LOW ENERGIES SO THAT ONLY ZERO NEUTRON ANGULAR MOMENTA ($L = 0$) INTERACTIONS ARE IMPORTANT. INTERFERENCE BETWEEN LEVELS IS NEGLECTED, HOWEVER, INTERFERENCE BETWEEN RESONANCE SCATTERING AND POTENTIAL SCATTERING IS INCLUDED. THERE ARE PROVISIONS FOR ADDING A CORRECTION CROSS SECTION OF THE FORM $1.0/\text{SQRT}(E)$ FOR LEVELS NOT EXPLICITLY CONSIDERED. ALSO THE CROSS SECTION IN THE WINGS OF A NEGATIVE ENERGY RESONANCE, $\text{CONSTANT}/(\text{SQRT}(E)*(E-E_0)**2)$ MAY BE ADDED.

247 FLANGE1 FLANGE1 OPERATES ON A TABULATED NEUTRON SCATTERING LAW, $S(\alpha, \beta)$, TO DEVELOP SEVERAL TYPES OF CROSS SECTION INFORMATION. THE VARIOUS CALCULATED CROSS SECTIONS NOW AVAILABLE ARE -

- SIGMA (E-PRIME TO E, MU)
- SIGMA (E-PRIME, MU)
- SIGNAL (E-PRIME TO E)
- SIGNAL (E-PRIME)
- M2 (E-PRIME), SECOND ENERGY TRANSFER MOMENT
- X1 SIGMAS BAR, AVERAGE SLOWING DOWN POWER
- SIGNAL (MAXWELLIAN AVERAGED)
- L REFERS TO THE LEGENDRE MOMENT, $L = 0, 1, 2, 3$ ARE AVAILABLE.
- M2 (MAXWELLIAN AVERAGED).

A TAPE OF ANGULAR CROSS SECTIONS IS PREPARED TO BE USED AS INPUT TO THE MULTIPLE SCATTERING CODE MUSE.

254 2PLUS THE 2PLUS CODE SOLVES THE PROBLEM OF THE SCATTERING OF CHARGED OR UNCHARGED NUCLEONS BY A NUCLEUS REPRESENTED BY A DEFORMED NUCLEAR POTENTIAL. THE MODEL ASSUMES THAT THE TARGET NUCLEUS HAS A $0+$ GROUND STATE AND A $2+$ FIRST EXCITED LEVEL, AND THE INTERACTION POTENTIAL HAS A QUADRUPOLE DEFORMATION. A HAUSER-FESHBACH COMPOUND NUCLEUS CALCULATION HAS BEEN INCLUDED. THE OUTPUT CONTAINS TOTAL, POTENTIAL ELASTIC, POTENTIAL INELASTIC ($2+$), REACTION, AND COMPOUND NUCLEUS CROSS SECTIONS AS WELL AS ELASTIC AND INELASTIC ANGULAR DISTRIBUTIONS.

263 GASKET GASKET CALCULATES THE THERMAL NEUTRON SCATTERING LAW, $S(\alpha, \beta)$, FOR A LARGE CLASS OF MODERATORS. PROVISION HAS BEEN MADE IN GASKET FOR THE FOLLOWING DYNAMICAL MODES OF THE SCATTERER -

- (1) FREE TRANSLATION (GAS)
- (2) DIFFUSIVE OR BROWNIAN MOTION
- (3) HARMONIC ISOTROPIC VIBRATIONS WITH CONTINUOUS FREQUENCY SPECTRUM
- (4) HARMONIC ANISOTROPIC VIBRATIONS WITH CONTINUOUS FREQUENCY SPECTRUM (AS APPLIED FOR INSTANCE TO GRAPHITE), AND
- (5) HARMONIC ISOTROPIC VIBRATIONS WITH DISCRETE FREQUENCY SPECTRUM.

289 GAKER THE GAKER CODE EVALUATES THE INELASTIC DOUBLE-DIFFERENTIAL NEUTRON SCATTERING CROSS SECTIONS FOR MODERATORS WITH PHONON SPECTRA WHICH CAN BE REPRESENTED AS SUMS OF DELTA-FUNCTIONS. IT IS BASED ON THE ORIGINAL MODEL FOR LIGHT WATER BY NELKIN, WHICH CONSISTED OF A TRANSLATOR, A HINDERED ROTATOR (TREATED AS AN ISOTROPIC OSCILLATOR), AND SEVERAL VIBRATIONAL OSCILLATORS. THE CODE HAS BEEN MODIFIED SEVERAL TIMES TO INCLUDE MORE OSCILLATORS AND TO TREAT ANISOTROPIC EFFECTS. FINAL ENERGY-INTEGRATED CROSS SECTIONS ARE ALSO CALCULATED.

292 PSEUDO
 STRUCTURED FROM AVERAGE NUCLEAR PROPERTIES IN THE RESOLVED RESONANCE REGION.

305 STRIP
 CAPTURE AND FISSION RESONANCE INTEGRALS ARE CALCULATED BY A FAST METHOD IN THE RESOLVED RESONANCE RANGE, TAKING EXPLICIT ACCOUNT OF OVERLAP AND INTERFERENCE BETWEEN RESONANCES IN A MIXTURE OF RESONANCE ABSORBERS. THE RESONANCE INTEGRALS ARE CALCULATED OVER ARBITRARY ENERGY BANDS. OPTIONALLY, THE NEUTRON FLUX AS A FUNCTION OF ENERGY IN ONE OR TWO REGIONS MAY BE PRINTED OUT.

308 JUPITOR1
 THIS IS A PROGRAM FOR PERFORMING COUPLED-CHANNEL CALCULATIONS TO EVALUATE THE CROSS SECTIONS FOR THE SCATTERING OF NUCLEAR PARTICLES BY VARIOUS COLLECTIVE NUCLEI.

323 MUFFLE
 THIS PROGRAM COMPUTES THE NEUTRON CROSS SECTIONS FOR A FISSIONABLE NUCLIDE IN WHICH ONE TO THREE REACTION CHANNELS ARE OPEN FOR THE FISSION PROCESS. PROVISION IS MADE FOR TWO INDEPENDENT SPIN STATES WITH INTERFERING LEVELS AS WELL AS A SET OF NON-INTERFERING LEVELS. THE CROSS SECTIONS ARE INTEGRATED OVER A SPECIFIED GROUP STRUCTURE TO YIELD RESONANCE INTEGRALS.

334 PEGGY
 PEGGY IS A LEAST SQUARES SEARCH PROGRAM WHICH ANALYZES, IN TERMS OF PHASE SHIFTS, THE ELASTIC SCATTERING OF SPIN ZERO AND SPIN ONE-HALF PARTICLES BY SPIN ZERO NUCLEI. REAL OR COMPLEX PHASE SHIFTS MAY BE USED WITH OR WITHOUT SPIN-ORBIT COUPLING. DIFFERENTIAL CROSS SECTION AND POLARIZATION ANGULAR DISTRIBUTIONS MAY BE ANALYZED EITHER SEPARATELY OR SIMULTANEOUSLY.

335 RAMES
 RAMES COMPUTES BOTH LOCAL AND NONLOCAL RADIAL INTEGRALS OF A VARIETY OF RADIAL OPERATORS USING SINGLE-PARTICLE WAVE FUNCTIONS WHICH ARE EIGENSTATES OF MOTION IN A WOODS-SAXON POTENTIAL WELL. THE OPERATORS CURRENTLY AVAILABLE ARE R^{*N} , $N=0,1,2,3,4,5$ AND THE DERIVATIVE WITH RESPECT TO X OF $1/(\exp(X)+1)$ WHERE $X=(R-R(0))/A(0)$.

B. SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS

6/68

33 GAM1/REP/UPDATE

THIS PROGRAM COMPUTES THE SLOWING-DOWN SPECTRUM IN EITHER THE P1 OR THE B1 APPROXIMATION USING 68 GROUPS OF NEUTRONS WITH A CONSTANT GROUP WIDTH $DU=0.25$. MULTIGROUP CONSTANTS ARE CALCULATED FOR UP TO 32 FAST GROUPS.

50 TEMPEST2

TEMPEST2 IS A NEUTRON THERMALIZATION PROGRAM BASED UPON THE WIGNER-WILKINS APPROXIMATION FOR LIGHT MODERATORS AND THE WILKINS APPROXIMATION FOR HEAVY MODERATORS. A MAXWELLIAN DISTRIBUTION MAY ALSO BE USED. THE MODEL USED MAY BE SELECTED AS A FUNCTION OF ENERGY. THE SECOND-ORDER DIFFERENTIAL EQUATIONS ARE INTEGRATED DIRECTLY RATHER THAN TRANSFORMED TO THE RICCATI EQUATION. THE PROGRAM PROVIDES MICROSCOPIC AND MACROSCOPIC CROSS SECTION AVERAGES OVER THE THERMAL NEUTRON SPECTRUM.

51 FCRM

THE FORM, OR FORTRAN-MUFT, PROGRAM IS A FOURIER TRANSFORM SLOWING-DOWN CODE. A LIBRARY TAPE CONTAINING 54-GROUP MICROSCOPIC CROSS SECTIONS, RESONANCE PARAMETERS, INELASTIC SCATTERING MATRICES, AND SOURCE SPECTRA IS USED TO GENERATE A 54-GROUP FLUX SPECTRUM AND FEW-GROUP CONSTANTS.

52 SAIL

THE MONOENERGETIC NEUTRON TRANSPORT EQUATION IS SOLVED USING THE DISCRETE SN METHOD FOR A ONE-DIMENSIONAL PLANE CELL. CELL PROPERTIES ARE COMPUTED.

53 S4 CYL GEOMETRY CELL CODE

THIS PROGRAM SOLVES THE ONE-DIMENSIONAL MONOENERGETIC BOLTZMANN EQUATION IN CYLINDRICAL GEOMETRY, USING THE S4 APPROXIMATION. IN ADDITION TO THE FLUX DISTRIBUTION, CELL-AVERAGED PARAMETERS ARE COMPUTED.

108 BAM

BAM COMPUTES THERMAL CONSTANTS, SPATIAL AND ENERGY DISTRIBUTIONS IN HETEROGENEOUS CYLINDRICAL CELLS BY ASSUMING SEPARABILITY OF SPACE AND ENERGY IN THE BOLTZMANN EQUATION.

113 ZCT

ZOT TAKES MULTIGROUP NEUTRON CROSS SECTION SETS IN THE SN FORMAT (SNG, DSN, ETC.) AND REDUCES THE NUMBER OF GROUPS (COLLAPSES) ACCORDING TO A GIVEN OR COMPUTED MULTIGROUP FLUX SPECTRUM. AVERAGE VELOCITIES FOR THE FEW-GROUP SET MAY BE GENERATED ON THE BASIS OF THE FLUXES AND VOLUMES FOR A GIVEN REACTOR CONFIGURATION.

119 QUICKIE QUICKIE SOLVES THE NEUTRON SLOWING DOWN AND THERMALIZATION PROBLEM IN INFINITE MEDIA BY INVERTING A SET OF SIMULTANEOUS MULTIGROUP EQUATIONS. THE CODE USES THE ULCER (ACC ABSTRACT 118) LIBRARY TAPE AND IS IN EXCELLENT AGREEMENT WITH ULCER FOR THOSE CASES WHERE BUCKLING IS KNOWN.

147 AILMOE AILMOE IS A MODIFIED FORM OF THE ANL ELMOE PROGRAM. THE FOURIER TRANSFORM OF THE FAST NEUTRON FLUX IS FOUND FOR A MIXTURE OF MODERATORS HEAVIER THAN HYDROGEN WITH THE MODERATOR SCATTERING LAW RIGOROUSLY ACCOUNTED FOR.

149 TYCHE3 TYCHE3 IS A MONTE CARLO CODE DESIGNED TO FIND THE SECOND, FOURTH AND SIXTH MOMENTS OF THE NEUTRON SLOWING DOWN DENSITY DISTRIBUTION IN AN INFINITE HOMOGENEOUS MEDIUM. NEUTRON WEIGHTS ARE USED TO AVOID THE TERMINATION OF A HISTORY BY ABSORPTION AND MINIMIZE THE RUNNING TIME. PROVISIONS ARE MADE FOR RESTART OF NON-CONVERGED PROBLEMS, GRAPHICAL DISPLAYS OF THE MOMENTS AND AVERAGE FISSION ENERGY AS A FUNCTION OF THE NUMBER OF SETS OF HISTORIES AND CALCULATION OF THE CORRECTION TO FLUX MOMENTS.

150 DANCORFF JR. THIS CODE EVALUATES MODERATOR SPACE CHORD DISTRIBUTION FUNCTIONS OF ZEROth AND FIRST ORDER, PLUS THEIR LINEAR, SQUARE, LOGARITHMIC AND EXPONENTIAL MOMENTS, FOR REGULAR AND IRREGULAR LATTICES OF CYLINDRICAL FUEL RODS CLAD WITH MATERIAL OF NEGLIGIBLE TOTAL CROSS SECTION. OF PARTICULAR SIGNIFICANCE FOR REACTOR DESIGN CALCULATIONS IS THE EXPONENTIAL MOMENT, OR DANCORFF CORRECTION, WHICH CAN BE CALCULATED EXACTLY IN INFINITE SQUARE AND HEXAGONAL LATTICES, IN CLUMPED SQUARE LATTICES WITH STRAIGHT OR CRUCIFORM WATER GAPS, OR IN CLUSTERS OF TWO, THREE, SEVEN, AND NINETEEN FUEL RODS.

160 SOPHIST1/2/5 SOPHIST1 CALCULATES TEMPERATURE-DEPENDENT MULTIGROUP ENERGY TRANSFER COEFFICIENTS FOR A MAXWELL GAS MODERATOR WITH ELASTIC, ISOTROPIC SCATTERING IN THE CENTER OF MASS SYSTEM. SOPHIST2 CALCULATES TEMPERATURE-DEPENDENT MULTIGROUP CROSS SECTIONS FOR A MAXWELL GAS. SOPHIST5 CALCULATES MULTIGROUP ENERGY TRANSFER MATRICES FOR ANISOTROPIC ELASTIC SCATTERING.

162 GRAVE GRAVE IS THE PROGRAM TO FORM GROUP-AVERAGED CROSS SECTIONS USING THE NEUTRON CROSS SECTION MASTER TAPE DEVELOPED UNDER THE ATOMICS INTERNATIONAL AUTOMATED CROSS SECTION PROGRAM. NINETEEN GROUP-AVERAGED PARAMETERS ARE OBTAINABLE AS FOLLOWS - SIGMA T, SIGMA EL, SIGMA N, SIGMA F, NUBAR, MUBAR, XIBAR, SIGMA IN, SIGMA N-ALPHA, SIGMA N-P, SIGMA N-2N, SIGMA A, SIGMA TR, SIGMA NON-EL, ALPHABAR, XI*SIGMA EL, NU*SIGMA F, MUBAR*SIGMA EL, AND SIGMA R. THE SPECTRUM IS CONSTRUCTED FROM A COMBINATION OF FISSION, E*-N, POWER SERIES, MAXWELLIAN OR INPUT SPECTRA.

178 SPARTA
SPARTA COMPUTES SPATIALLY-AVERAGED DOPPLER COEFFICIENTS AND SPATIALLY-AVERAGED DOPPLER REACTIVITY CHANGES.

184 THERMOS(ANL)
THIS ARGONNE-REVISED VERSION LIKE THE ORIGINAL THERMOS CODE DEVELOPED BY H. HONECK OF BROOKHAVEN NATIONAL LABORATORY COMPUTES THE SCALAR THERMAL NEUTRON SPECTRUM AS A FUNCTION OF POSITION IN A LATTICE BY SOLVING THE INTEGRAL TRANSPORT EQUATION WITH ISOTROPIC SCATTERING. ONE-DIMENSIONAL SLAB OR CYLINDRICAL GEOMETRY MAY BE USED. AS OUTPUT THE CODE SUPPLIES FLUX-AVERAGED VALUES OF SIGMA A, SIGMA F, NU SIGMA F, SIGMA S, AND D FOR THE CELL COMPOSITION AND THE VALUES OF SIGMA A, SIGMA F, NU SIGMA F, SIGMA S, AND SIGMA TR FOR THE ISOTOPIC CONSTITUENTS.

185 GAMTEC2
GAMTEC2 GENERATES MULTIGROUP CONSTANTS IN THE ENERGY RANGE FROM 0 TO 10 MEV FOR EITHER HOMOGENEOUS MIXTURES OR HETEROGENEOUS ARRAYS CONSISTING OF CYLINDERIZED LATTICE CELLS. THE THERMAL GROUP CONSTANTS ARE AVERAGED OVER EITHER (1) WIGNER-WILKINS LIGHT MODERATOR SPECTRUM, (2) WILKINS HEAVY MODERATOR SPECTRUM, OR (3) A MAXWELLIAN DISTRIBUTION. FOR HETEROGENEOUS ARRAYS THE SPATIAL THERMAL FLUX IS CALCULATED BY A MONOENERGETIC P3 APPROXIMATION. FOR EPITHERMAL ENERGIES, THE SLOWING-DOWN DISTRIBUTION IS DESCRIBED BY EITHER A B1 OR P1 APPROXIMATION TO THE BOLTZMANN EQUATION. RESONANCE ABSORPTION AND FISSION ARE TREATED BY THE ADLER-NORDHEIM METHOD. AN IMPROVED METHOD OVER THAT IN GAM-I FOR AVERAGING THE RESONANCE ABSORPTION CONTRIBUTION TO THE MULTIGROUP CONSTANTS IS INCLUDED. FUEL LUMPING EFFECTS ON THE FAST FISSION OF U238 AND TH232 ARE TREATED BY AN N-FLIGHT COLLISION PROBABILITY TECHNIQUE. GROUP CONSTANTS ARE PUNCHED ON CARDS IN HFN (DIFFUSION CODE) AND DTF (SN TRANSPORT CODE) FORMATS.

195 FORTRAN FMC-N/FMC-G
THE FLEXIBLE MONTE CARLO PROGRAMS FMC-N AND FMC-G WERE ORIGINALLY WRITTEN IN ASSEMBLY LANGUAGE FOR THE IBM704 BY GE-ANP. THESE PROGRAMS HAVE BEEN CONVERTED TO FORTRAN FOR THE CDC16048 BY PRATT AND WHITNEY. THE PROGRAMS APPLY MONTE CARLO METHODS TO SIMULATE NEUTRON AND GAMMA RAY LIFE HISTORIES, RESPECTIVELY, IN A SOURCE-SHIELD CONFIGURATION. AS THE NAMES IMPLY, THE CODES ARE DESIGNED FOR FLEXIBILITY IN THE GEOMETRICAL, MATERIAL, NUCLEAR, AND SOURCE DESCRIPTIONS OF SOURCE-SHIELD CONFIGURATIONS AND VARIANCE REDUCTION TECHNIQUES.

201 EPITHERMOS THE EPITHERMOS CODE IS A MODIFICATION OF THE THERMOS THERMAL TRANSPORT THEORY CODE OF HONECK. THE ESSENTIAL DIFFERENCE BETWEEN THE CODES IS IN THE LIBRARY PREPARATION SUBROUTINE BASK. THE EPITHERMOS CODE IS DESIGNED TO COMPUTE THE NEUTRON DENSITY ABOVE 0.7849 EV, THE MAXIMUM ENERGY OF THE USUAL THERMOS CALCULATION. EPITHERMOS COMPUTES THE SCALAR NEUTRON DENSITY AS A FUNCTION OF POSITION AND SPEED IN A ONE-DIMENSIONAL SLAB OR CYLINDRICAL SYSTEM. THE SCATTERING MODEL IS ARBITRARY IN THE THERMAL RANGE AND THE BUILT-IN FREE GAS MODEL IS USED IN THE EPITHERMAL RANGE.

202 MCS THE MCS CODE DETERMINES THE SPATIAL DISTRIBUTION OF SOME NUCLEAR REACTION FOR A GIVEN NEUTRON SOURCE IN A GIVEN CONFIGURATION OF MATERIALS. THE MONTE CARLO ESTIMATE OF THE SOLUTION CONSISTS OF PICKING A SAMPLE OF NEUTRONS FROM THE GIVEN SOURCE AND FOLLOWING EACH NEUTRON THROUGH A SEQUENCE OF SURFACE CROSSINGS AND COLLISIONS UNTIL THE NEUTRON EITHER ESCAPES OR IS NO LONGER OF INTEREST TO THE SOLUTION. THE DESIRED FLUX OR COLLISION DENSITY IS ACCUMULATED FOR EVERY NEUTRON OF THE SAMPLE, AND THE SAMPLE SIZE INCREASED UNTIL RESULTS OF SUFFICIENT STATISTICAL SIGNIFICANCE ARE OBTAINED.

204 AGN-GAM AGN-GAM CALCULATES THE SUBGROUP FLUXES AND CURRENT TERMS FROM A SOLUTION OF THE P1 OR B1 EQUATIONS. THE P1 EQUATIONS INCORPORATE BOTH VOLUME AND SURFACE SOURCES, ALLOWING SPECTRAL CALCULATIONS IN REFLECTOR REGIONS. OTHER SPECTRAL OPTIONS ARE FLUX KNOWN, CURRENT TERM CALCULATED AND BOTH FLUX AND CURRENT TERMS KNOWN. THE AGE FOR THE MATERIAL IS OBTAINED FROM A SECOND MOMENTS CALCULATION. THE 75 SUBGROUPS USED IN THE CALCULATIONS ARE ESTABLISHED AS QUARTER LETHARGY GROUPS FROM 10 MEV TO 0.07EV. ELASTIC SCATTERING MATRICES MAINTAIN THE ENERGY-ANGLE CORRELATION BY INCLUDING BOTH P0 AND P1 TERMS. ISOTROPIC SCATTERING MATRICES ARE INCLUDED FOR INELASTIC AND (N,2N) PROCESSES. THE METHOD OF ADLER, HINMAN AND NORDHEIM IS USED TO CALCULATE RESONANCE ABSORPTION AND FISSION CROSS SECTIONS. MULTIGROUP CONSTANTS ARE GENERATED BY SPECTRAL AVERAGING OVER THE SUBGROUPS.

213 RIFF-RAFF THE PROGRAM COMPUTES RESONANCE INTEGRALS FOR ABSORBERS IN A ROD IN A TWO-REGION CIRCULARIZED CELL FOR ABOVE THERMAL ENERGIES. THE FLUX DISTRIBUTION IN THE CELL IS ALSO COMPUTED AND CAN BE PRINTED. THE METHOD OF SOLUTION IS BASED UPON THE FOLLOWING ASSUMPTIONS - ISOTROPIC SCATTERING IN THE LABORATORY SYSTEM, CONSTANT TOTAL CROSS SECTION IN THE MODERATOR AND ISOTROPIC NEUTRON FLUXES ENTERING AND LEAVING THE ROD.

235 GAMMA-P (NMP468) CROSS SECTIONS FOR THE PRODUCTION OF GAMMA RAYS BY NEUTRON RADIATIVE CAPTURE, BY NEUTRON INELASTIC SCATTERING, AND BY NEUTRON-INDUCED FISSIONING ARE COMPUTED AS A FUNCTION OF NEUTRON LETHARGY GROUPS AND PHOTON ENERGY GROUPS.

237 BOUNCE THE BOUNCE CODE DETERMINES THE ONE-GROUP THERMAL NEUTRON FLUX DISTRIBUTION WITHIN THE PIN BUNDLE OF A MULTIPLE-PIN FUEL ELEMENT. ALL SCATTERING WITHIN THE PINS IS CONSIDERED TO BE ISOTROPIC. ELEMENTS WITH 6, 7, 12, 13, 18, AND 19 FUEL PINS, WHERE EACH PIN IS COMPOSED OF A CENTRAL FUEL PELLET SURROUNDED BY A CLADDING MATERIAL, MAY BE TREATED. THE COOLANT REGION AROUND THE PINS IS ASSUMED TO CONTAIN A VOID.

243 AGN-SIGMA AGN-SIGMA CALCULATES THE LEGENDRE COMPONENTS OF THE MULTIGROUP TRANSFER MATRICES $\Sigma(L, G \text{ TO } G+N)$ FOR FAST NEUTRONS. REACTIONS CONSIDERED ARE ELASTIC SCATTERING, INELASTIC SCATTERING (LEVEL EXCITATION AND THE EVAPORATION MODEL), AND THE FOLLOWING FIVE DECAY MODES FOR THE $(N, 2N)$ REACTION -- $A(N, N1)A^*(N2)(A-1)^*$, 3- AND 4- BODY PHASE SPACE MODEL, EVAPORATION MODEL, AND THE CLUSTER MODEL, WHERE A^* IS THE RECOIL NUCLEUS. ALL NUCLEAR LEVELS INVOLVED IN THE TRANSITIONS ARE DISCRETE. THE CODE MAY ALSO BE USED TO CALCULATE GROUP-AVERAGED CROSS SECTIONS AS WELL AS TO MANIPULATE, E.G., ADD, MULTIPLY, ETC., THE OUTPUT MATRICES. THE NEUTRON SPECTRUM MAY BE A COMBINATION OF FISSION AND $1/E$ OR ARBITRARY INPUT DATA.

249 LASER LASER IS BASED ON MODIFIED VERSIONS OF THE SLOWING-DOWN PROGRAM MUFT AND THE THERMALIZATION TRANSPORT THEORY PROGRAM THERMOS, AND PERFORMS A CALCULATION OF THE NEUTRON SPECTRUM IN A UNIFORM LATTICE MADE UP OF CYLINDRICAL RODS, CLADDING, AND SURROUNDING MODERATOR. THE THERMAL CUTOFF IN LASER IS 1.855 EV. THE PROGRAM PERFORMS A BURNUP CALCULATION FOR THE LATTICE. THE SPATIAL DISTRIBUTION OF BURNUP WITHIN THE FUEL RODS IS EXPLICITLY CALCULATED. THE PROGRAM WILL, AT OPTION, ACCOUNT FOR ALL NON-LINEARITIES AND MUTUAL CONNECTIONS IN THE SYSTEM OF BURNUP EQUATIONS. THIS CALCULATION ACCOUNTS FOR THE VARIATION OF THE NEUTRON FLUX IN SPACE AND ENERGY DURING EACH TIME-STEP. A BUCKLING AND A BORON POISON SEARCH (CRITICALITY SEARCH) ARE PROVIDED AS OPTIONS. OUTPUT INCLUDES EDITS IN THE ENERGY RANGE ZERO LESS THAN OR EQUAL TO E LESS THAN OR EQUAL TO 0.625 EV.

257 REAX REAX CALCULATES EPITHERMAL FLUX, ACTIVITIES AND CROSS SECTIONS AS A FUNCTION OF RADIUS AND ENERGY FOR A CONSTANT TEMPERATURE FUEL ROD IMMERSSED IN A HOMOGENEOUS MEDIUM.

276 AVCID AVOID COMPUTES THE EQUIVALENT DIFFUSION COEFFICIENT AND LOSS CROSS SECTION OF AN ANNULAR VOID IN A CYLINDRICAL REACTOR AND THE RADIAL FLUX DISTRIBUTION IN THE VOID.

277 HAMMER/LITHE/HELP/LIBCON HAMMER PERFORMS INFINITE LATTICE, ONE-DIMENSIONAL CELL MULTIGROUP CALCULATIONS, FOLLOWED (OPTIONALLY) BY ONE-DIMENSIONAL, FEW-GROUP, MULTIREGION REACTOR CALCULATIONS WITH NEUTRON BALANCE EDITS.

279 LEOPARD/AITP1/AITP3/SPCTS LEOPARD IS A UNIT CELL HOMOGENIZATION AND SPECTRUM GENERATION (MUFT-SOFOCATE) PROGRAM WITH A FUEL DEPLETION OPTION.

R280 M0807 M0807 SOLVES THE TWO-DIMENSIONAL FIXED SOURCE DIFFUSION EQUATION FOR THE ABSORPTION AND REMOVAL MACROSCOPIC CROSS SECTIONS REQUIRED TO YIELD A SPECIFIED REACTION RATE DISTRIBUTION.

281 RABBLE/WLIB/FLAT RABBLE COMPUTES EFFECTIVE CROSS SECTIONS FOR ABOVE THERMAL ENERGIES BASED ON RESOLVED SINGLE-LEVEL RESONANCE PARAMETERS FOR INFINITE HOMOGENEOUS OR HETEROGENEOUS SYSTEMS.

R285 RESQ2/RESQ0/DBF1 RESQ2 CALCULATES THE RESONANCE INTEGRAL IN A TWO-DIMENSIONAL, HEXAGONAL SYSTEM CONSISTING OF FUEL, CLAD AND WATER WITH A REFLECTING BOUNDARY CONDITION.

291 HEXSCAT HEXSCAT CALCULATES P_0 THROUGH P_3 COMPONENTS OF THE POLYCRYSTALLINE COHERENT ELASTIC NEUTRON SCATTERING CROSS SECTION PER NUCLEUS FOR A HEXAGONAL LATTICE. THE CODE AVERAGES POINT VALUES OVER INPUT GROUP BOUNDARIES TO GIVE SMOOTHED GROUP CROSS SECTIONS.

298 GGC4

THE GGC4 PROGRAM SOLVES THE MULTIGROUP SPECTRUM EQUATIONS WITH SPATIAL DEPENDENCE REPRESENTED BY A SINGLE POSITIVE INPUT BUCKLING. BROAD GROUP CROSS SECTIONS (SHIELDED OR UNSHIELDED) ARE PREPARED FOR DIFFUSION AND TRANSPORT CODES BY AVERAGING WITH THE CALCULATED SPECTRA OVER INPUT-DESIGNATED ENERGY LIMITS. THE CODE IS DIVIDED INTO THREE MAIN PARTS. A FAST (GAM) SECTION WHICH COVERS THE ENERGY RANGE FROM 14.9 MEV TO 0.414 EV, A THERMAL (GATHER) SECTION WHICH COVERS THE ENERGY RANGE FROM 0 TO 2.38 EV, AND A COMBINING (COMBO) SECTION WHICH COMBINES FAST AND THERMAL CROSS SECTIONS INTO SINGLE SETS. BASIC NUCLEAR DATA FOR THE FAST SECTION WHICH CONSISTS OF FINE GROUP-AVERAGED CROSS SECTIONS AND RESONANCE PARAMETERS IS READ OFF A DATA TAPE. THE FINE GROUP ABSORPTION AND FISSION CROSS SECTIONS MAY BE ADJUSTED BY PERFORMING A RESONANCE INTEGRAL CALCULATION. UTILIZING A FISSION SOURCE AND AN INPUT BUCKLING, THE CODE SOLVES THE P1, B1, B2, OR B3 APPROXIMATION TO OBTAIN THE ENERGY-DEPENDENT FAST SPECTRUM. TWO OR SIX SPATIAL MOMENTS OF THE SPECTRUM (DUE TO A PLANE SOURCE) MAY ALSO BE EVALUATED. INSTEAD OF PERFORMING A SPECTRUM CALCULATION, THE USER MAY ENTER THE LEGENDRE COMPONENTS OF THE ANGULAR FLUX DIRECTLY. FOR AS MANY INPUT-DESIGNATED BROAD GROUP STRUCTURES AS DESIRED, THE CODE CALCULATES AND SAVES (FOR THE COMBINING SECTION) SPECTRUM-WEIGHTED AVERAGES OF MICROSCOPIC AND MACROSCOPIC CROSS SECTIONS AND TRANSFER ARRAYS. SLOWING DOWN SOURCES ARE CALCULATED AND SAVED FOR USE IN THE LOWER ENERGY RANGE. GIVEN BASIC NUCLEAR DATA, THE THERMAL SECTION OF GGC4 DETERMINES A THERMAL SPECTRUM BY EITHER READING IT AS INPUT, BY CALCULATING A MAXWELLIAN SPECTRUM FOR A GIVEN TEMPERATURE, OR BY AN ITERATIVE SOLUTION OF THE P3, B0, P1, OR B1 EQUATIONS FOR AN INPUT BUCKLING. TIME MOMENTS OF THE TIME AND ENERGY-DEPENDENT DIFFUSION EQUATIONS ARE CALCULATED (AS AN OPTION) USING THE INPUT BUCKLING TO REPRESENT LEAKAGE. BROAD GROUP CROSS SECTIONS ARE PREPARED BY AVERAGING FINE GROUP CROSS SECTIONS OVER THE CALCULATED SPECTRA. BROAD GROUP STRUCTURES ARE READ AS INPUT. THE COMBINING SECTION OF GGC4 TAKES THE BROAD GROUP-AVERAGED CROSS SECTIONS FROM THE FAST AND THERMAL PORTIONS OF GGC4 AND FORMS MULTIGROUP CROSS SECTION TABLES. THESE TABLES ARE PREPARED IN STANDARD FORMATS FOR TRANSPORT OR DIFFUSION THEORY CALCULATIONS. IN ADDITION, IT IS POSSIBLE TO USE THE COMBINING SECTION TO PRODUCE MIXTURES NOT USED IN THE SPECTRUM CALCULATION OR TO COMBINE THE RESULTS OF DIFFERENT FAST AND THERMAL SECTION CALCULATIONS AND SO ON.

306 FCC4

FCC4 IS A MULTIPURPOSE DATA MANIPULATION CODE FOR USE IN FAST REACTOR ANALYSIS. THE CODE CAN BE USED TO - (A) COMPUTE RESONANCE-SHIELDED CROSS SECTIONS USING DATA IN THE RUSSIAN FORMAT (SHIELDING FACTORS AND INFINITE-DILUTION CROSS SECTIONS), (B) COMPUTE MULTIGROUP FUNDAMENTAL-MODE FLUX AND ADJOINT FLUX, (C) COMPUTE AND PUNCH GROUP-COLLAPSED MICROSCOPIC OR MACROSCOPIC CROSS SECTIONS IN THE DTF FORMAT, (D) COMPUTE FUEL BURNUP AT CONSTANT FLUX OR POWER DENSITY.

307 HWOCR-SAFE HWOCR-SAFE IS A MONTE CARLO THERMAL REACTOR ANALYSIS PROGRAM DESIGNED FOR USE WITH PROPOSED HWOCR LATTICE CONFIGURATIONS. IT IS A BENCHMARK TOOL TO CHECK MULTIGROUP DIFFUSION AND TRANSPORT CALCULATIONS AND TO EVALUATE THE EFFECT OF THEIR USE OF GEOMETRIC APPROXIMATIONS.

316 GAF/GAR THE PROBLEM IS TO OBTAIN VERY DETAILED NEUTRON FLUX AND CURRENT DISTRIBUTIONS AS FUNCTIONS OF ENERGY CONSIDERING EXPLICITLY THE POSSIBLE OVERLAP EFFECTS BETWEEN RESONANCES OF A RESONANCE ABSORBER AND OF MIXTURES OF RESONANCE ABSORBERS AND TO USE THESE DISTRIBUTIONS TO PREPARE GROUP-AVERAGED CROSS SECTIONS AND TRANSFER ARRAYS FOR USE IN FAST REACTOR ANALYSES.

C. STATIC DESIGN STUDIES

6/68

7 FIRM FIRM SOLVES THE TWO-DIMENSIONAL SN APPROXIMATION IN FINITE CYLINDRICAL GEOMETRY. IT IS A FORTRAN ADAPTATION OF THE LOS ALAMOS TDC PROGRAM WRITTEN BY BENGT CARLSON AND CLARENCE LEE. FIRM IS LIMITED TO A MAXIMUM OF SIX GROUPS AND THE S2, S4 OR S6 APPROXIMATION.

9 FIRE2 FIRE2 SOLVES THE ONE-DIMENSIONAL MULTIGROUP AGE-DIFFUSION EQUATIONS FOR SLAB (SYMMETRIC AND ASYMMETRIC), CYLINDER, AND SPHERE GEOMETRIES. AN EXTENSION OF THE ORIGINAL LOS ALAMOS FIRE PROGRAM WRITTEN BY F. W. BRINKLEY AND C. B. MILLS, IT CONTAINS A 34-GROUP LIBRARY OF 10 ELEMENTS.

18 2DXY THE 2DXY PROGRAM SOLVES THE HOMOGENEOUS OR INHOMOGENEOUS MULTIGROUP EQUATIONS IN X-Y GEOMETRY, USING THE SN TRANSPORT EQUATION APPROXIMATION. VACUUM, SURFACE SOURCE, OR REFLECTING BOUNDARY CONDITIONS ARE AVAILABLE AS OPTIONS. IN THE HOMOGENEOUS CASE THE USER MAY REQUEST THE COMPUTATION OF REACTIVITY, REACTOR PERIOD, CRITICAL CONCENTRATIONS OF SOME COMPOSITION, OR THE CRITICAL THICKNESS OF A ZONE.

28 FOG THE FOG PROGRAM SOLVES THE ONE-DIMENSIONAL FEW-GROUP DIFFUSION EQUATIONS IN ANY OF THREE GEOMETRIES SLAB, CYLINDER, OR SPHERE. PROVISIONS ARE MADE FOR CALCULATING THE FLUX, THE ADJOINT FLUX, AND VARIOUS CRITICALITY SEARCHES AS WELL AS A BUCKLING ITERATION AND AUTOMATIC TSCHEBYSCHIEFF POLYNOMIAL COMPUTATION OF SOURCE EXTRAPOLATION FACTORS.

29 AIM6 AIM6 SOLVES THE ONE-DIMENSIONAL MULTIGROUP DIFFUSION EQUATIONS UTILIZING A MICROSCOPIC CROSS SECTION LIBRARY. ANY OF THREE GEOMETRIES ARE AVAILABLE SLAB, CYLINDER OR SPHERE. CRITICALITY SEARCHES ARE PROVIDED INCLUDING A CONCENTRATION SEARCH ON ONE OR TWO ELEMENTS. HOMOGENEOUS AND INHOMOGENEOUS PROBLEMS MAY BE SOLVED WITH A VARIETY OF BOUNDARY CONDITION OPTIONS.

30 PERT PERT IS A PERTURBATION PROGRAM DESIGNED FOR USE WITH THE AIM6 AND FOG PROGRAMS. PUNCHED CARD OUTPUT FROM THESE CODES IS USED AS INPUT TO PERT. USING CROSS SECTION DATA, FLUXES, AND ADJOINT FLUXES, THE RELATIVE CHANGE IN KEFF CAN BE CALCULATED. CROSS SECTIONS MAY BE WEIGHTED WITH THE ADJOINT FLUX AND/OR DIRECT FLUX. THE NEUTRON LIFETIME FOR THE DELAY GROUPS MAY ALSO BE DETERMINED.

32 WHIRLAWAY WHIRLAWAY SOLVES THE TWO-GROUP, THREE-DIMENSIONAL, NEUTRON DIFFUSION EQUATIONS IN X-Y-Z GEOMETRY.

39 EQUIPOISE3 EQUIPOISE3 SOLVES THE TWO-GROUP, TWO-DIMENSIONAL, NEUTRON DIFFUSION EQUATIONS IN CYLINDRICAL OR SLAB GEOMETRY.

40 20GRAND 20GRAND SOLVES THE FEW-GROUP, TWO-DIMENSIONAL, NEUTRON DIFFUSION EQUATIONS IN CYLINDRICAL OR SLAB GEOMETRY.

59 MIST MIST OBTAINS THE SOLUTION TO THE ONE-DIMENSIONAL BOLTZMANN EQUATION IN SLAB GEOMETRY. THE NUMERICAL APPROXIMATION USED IS A LINEAR ONE WHICH CAN BE DESCRIBED AS AN EXTENSION AND GENERALIZATION OF THE SN APPROXIMATION. THE EQUATIONS ARE FORMULATED IN TERMS OF A DOUBLE SN APPROXIMATION. THE BOUNDARY CONDITIONS FOR EACH GROUP MAY BE INDEPENDENTLY SPECIFIED AND PERMIT VERY GENERAL SPECIFICATIONS WITH RESPECT TO -

- (A) PERFECT MIRROR REFLECTION OR SYMMETRY, BY INPUT OF MIRROR ALBEDOS,
- (B) ANISOTROPIC DIFFUSE SOURCES, BY INPUT OF LEGENDRE POLYNOMIAL COEFFICIENTS UP TO $\lambda = 9$, OR A SHORT TABLE DESCRIBING A KNOWN ANGULAR DISTRIBUTION OF THE FLUX,
- (C) ISOTROPIC (LAMBERT SURFACE) REFLECTION.

ISOTROPIC VOLUME SOURCES IN EACH GROUP MAY ALSO BE INDEPENDENTLY SPECIFIED. THE SCATTERING FROM ONE GROUP TO ANOTHER IS ASSUMED TO BE ISOTROPIC BUT THE SCATTERING FUNCTION WITHIN EACH GROUP CAN BE A SECOND-ORDER LEGENDRE POLYNOMIAL SERIES.

75 GE-HAPO-S13 PROGRAM S CONSTRUCTS BILINEARLY COUPLED TIME-VARIANT MULTIENERGY NEUTRON-AND-PHOTON TRANSPORT AND NUCLIDE-TRANSMUTATION FIELDS HAVING SLAB, CYLINDRICAL, OR SPHERICAL SYMMETRY. ASSURANCE OF UNBIASED CONVERGENCE IS PROVIDED BY USE OF A DUAL ADJOINT-AND-FLUX LOOP CONSTRUCTED IN PRECISE CORRESPONDENCE WITH THE PHYSICS OF SUCCESSIVE FREE FLIGHTS.

87 EQUIPOISE3A EQUIPOISE-3A SOLVES THE TWO-DIMENSIONAL TWO-GROUP DIFFUSION EQUATIONS IN CYLINDRICAL OR SLAB GEOMETRY. IT IS A SLIGHTLY REVISED VERSION OF EQUIPOISE3 (ACC ABSTRACT 39). IN ADDITION TO THE STANDARD OUTPUT, A PICTURE IS PRINTED OF THE MATERIAL ARRANGEMENT IN THE REACTOR. IF THE ADJOINT FLUX OPTION IS USED, THE PROMPT NEUTRON LIFETIME IS CALCULATED AND PRINTED, WITH THE REACTIVITY PER UNIT CHANGE IN EACH GROUP CONSTANT IN EACH REGION OF THE REACTOR.

103 CRAM CRAM IS A PROGRAM TO SOLVE THE MULTIGROUP DIFFUSION EQUATIONS IN TWO-DIMENSIONS (R-Z, X-Y, OR R-THETA GEOMETRY), OR IN ONE-DIMENSION (SLAB, CYLINDRICAL, OR SPHERICAL GEOMETRY). NEUTRONS MAY SCATTER FROM ANY GROUP TO ANY OTHER. REAL, ADJOINT, AND SOURCE-TYPE PROBLEMS ARE ALL SOLVABLE. THE PROGRAM WILL COMPUTE THE K-EFFECTIVE OF THE SYSTEM OR ALTERNATIVELY SEARCH FOR CRITICALITY BY MOVING SPATIAL BOUNDARIES, VARYING MATERIAL COMPOSITIONS, OR VARYING TRANSVERSE BUCKLING.

118 ULCER ULCER IS A MULTIGROUP, ONE-DIMENSIONAL DIFFUSION EQUATION CODE WITH UPSCATTER BASED ON FAIM (ACC ABSTRACT 120). ULCER DIFFERS FROM FAIM IN THAT -

- (A) UPSCATTERING IS INCLUDED,
- (B) DOWNSCATTER TO ALL LOWER GROUPS,
- (C) MICROSCOPIC CROSS SECTIONS ARE ON TAPE,
- (D) PROVISION FOR MULTIPLE FISSION SPECTRA,
- (E) RESTART DUMP,
- (F) FEW-GROUP REDUCTION, AND
- (G) SPECTRUM COMPUTATION AND GRAPHICAL DISPLAY.

120 FAIM/FAIMOS FAIM IS A MULTIGROUP, ONE-DIMENSIONAL DIFFUSION EQUATION PROGRAM BASED ON AIM6 (ACC ABSTRACT 29). THE PRINCIPAL FEATURES ARE -

- (A) THREE GEOMETRIES,
- (B) CALCULATION OF FLUXES AND MULTIPLICATION FACTOR,
- (C) ONE-ITERATION PROBLEMS,
- (D) CHOICE OF ONE OF FIVE SETS OF BOUNDARY CONDITIONS AT BOTH BOUNDARIES,
- (E) CRITICALITY SEARCHES ON TRANSVERSE BUCKLING, HOMOGENEOUS POISON, CRITICAL RADIUS, ONE, TWO, OR THREE ELEMENT CONCENTRATION, LOCATION OF POISON REGION BOUNDARY, LOCATION OF A FUEL REGION BOUNDARY,
- (F) ADJOINT FLUX CALCULATION, AND
- (G) EXTENSIVE DATA EDIT.

FAIMOS IS A MODIFIED VERSION OF FAIM. THREE GENERAL MODIFICATIONS WERE MADE -

- (A) THE MICROSCOPIC CROSS SECTION LIBRARY AND ITS ASSOCIATED SUBROUTINES WERE REMOVED. AS A RESULT, OPTIONS REQUIRING THE USE OF MICROSCOPIC CROSS SECTIONS ARE NOT AVAILABLE.
- (B) FAIM RAN AS A CHAIN JOB. OVERLAYS ARE NOT USED BY FAIMOS.
- (C) THE PROGRAM LANGUAGE WAS CONVERTED FROM FORTRAN II TO FORTRAN IV(H). THIS MODIFICATION MADE NECESSARY A MINOR CHANGE IN THE DATA INPUT FORMAT.

132 W-DSN W-DSN SOLVES THE DISCRETE SN EQUATIONS IN CYLINDRICAL GEOMETRY. THE EIGENVALUE OPTION IS REACTIVITY (KEFF) ONLY. VOLUME DISTRIBUTED SOURCES ARE ALLOWED, BUT NO SURFACE SOURCES.

136 HERESY1/KERNEL HERESY1 CALCULATES THE REACTIVITY, THERMAL UTILIZATION, RESONANCE ESCAPE PROBABILITY, RELATIVE ROD ABSORPTIONS AND POWER DISTRIBUTION IN HETEROGENEOUS REACTORS HAVING TWO SPATIAL DIMENSIONS. HERESY1 CAN BE USED FOR NON-UNIFORM LATTICES, LATTICES WITH MANY TYPES OF FUEL AND CONTROL RODS, AND SPIKED AND SEEDED REACTORS. FISSIONS ARE ASSUMED TO OCCUR ONLY AT THERMAL ENERGY. RESONANCE ABSORPTIONS ARE LUMPED INTO ONE EQUIVALENT RESONANCE. THE MODERATOR IS ASSUMED TO BE INFINITE IN THE RADIAL DIRECTION, AND RODS ARE TREATED AS LINE SOURCES AND SINKS. THE ROD PARAMETERS ARE INDEPENDENT OF THE INTER-ROD SEPARATION DISTANCES. SLOWING-DOWN KERNEL FUNCTIONS MAY BE OF ANY TYPE - AGE THEORY, TRANSPORT THEORY, OR EMPIRICAL. A SELF-CONSISTENT PROCEDURE CAN BE USED WHICH EFFECTIVELY CANCELS OUT ANY ERRORS IN THE KERNEL FUNCTIONS.

144 DTF THE DTF PROGRAM IS A ONE-DIMENSIONAL MULTIGROUP PROGRAM FOR SOLVING THE NEUTRON TRANSPORT EQUATION. THE PROGRAM CAN DETERMINE THE REGULAR OR ADJOINT SOLUTION FOR SLAB, CYLINDRICAL, OR SPHERICAL GEOMETRY. ISOTROPIC OR A FORM OF LINEAR ANISOTROPIC SCATTERING MAY BE CONSIDERED. VARIOUS BOUNDARY CONDITIONS ARE ALLOWED SO THAT CELLS OR TIME-DEPENDENT SOLUTIONS MAY BE OBTAINED USING FINITE AS WELL AS INFINITE CONFIGURATIONS IN THE CASE OF SLABS OR CYLINDERS. THE PROGRAM ALSO CONTAINS A NUMBER OF SEARCH OPTIONS WHEREBY ONE CAN VARY DIMENSIONS OR CONCENTRATIONS TO ARRIVE AT A PREDETERMINED EIGENVALUE. DISTRIBUTED OR SHELL SOURCES MAY BE SPECIFIED AT ANY POSITION WITHIN THE CONFIGURATION. AS OUTPUT THE PROGRAM SUPPLIES THE EIGENVALUE, ANGULAR FLUXES, TOTAL FLUXES, FISSION DISTRIBUTIONS, AND OTHER QUANTITIES. A LIBRARY OF CROSS SECTIONS IS AVAILABLE ON MAGNETIC TAPE. CROSS SECTIONS MAY BE READ FROM THIS LIBRARY TAPE AND/OR FROM CARDS.

148 TOPIC TOPIC SOLVES THE ONE-DIMENSIONAL BOLTZMANN EQUATION IN CYLINDRICAL GEOMETRY WITH UP TO SIX ENERGY GROUPS, 240 SPACE POINTS, 40 REGIONS, AND ANISOTROPIC (P1) SCATTERING.

THE BOUNDARY CONDITIONS FOR EACH GROUP CAN BE INDEPENDENTLY SPECIFIED AND THE FLEXIBILITY OF THE SPECIFICATIONS PERMIT -

- (A) PERFECT MIRROR REFLECTION OR SYMMETRY,
- (B) ISOTROPIC REFLECTION (LAMBERT SURFACE REFLECTION), AND
- (C) ANISOTROPIC DIFFUSE SOURCES BY MEANS OF EITHER
A P1 LEGENDRE SERIES OR A SHORT TABLE OF POINT VALUES FOR
THE ANGULAR FLUX.

INDEPENDENT SPECIFICATION OF ISOTROPIC FIXED VOLUME SOURCES FOR EACH GROUP IS ALSO ALLOWED.

AS IMPLIED, BOTH HOMOGENEOUS AND INHOMOGENEOUS PROBLEMS ARE SOLVED, AND FISSIONS CAN OCCUR IN EITHER TYPE OF PROBLEM.

151 DTF2/ANISN THE MULTIGROUP, ONE-SPACE DIMENSION NEUTRON TRANSPORT EQUATION IS SOLVED. ISOTROPIC OR LINEAR ANISOTROPIC SCATTERING IS PERMITTED BETWEEN ALL GROUPS AND A DIFFUSION SOLUTION MAY BE OBTAINED FOR ANY OR ALL GROUPS. HIGH-ORDER ANISOTROPIC SCATTERING PROBLEMS (PL) CAN BE RUN USING ANISN, THE 360 VERSION OF DTF2. WHITE/GREY BOUNDARY CONDITIONS ARE AVAILABLE, AND AN ALBEDO CAN BE SPECIFIED FOR EACH GROUP. A VOID STREAMING CORRECTION IS INCLUDED. A COMPLETE SHELL SOURCE DESCRIPTION BY GROUP, POSITION, AND ANGLE IS AVAILABLE. GRAPHICAL DISPLAY (CRT) FEATURES ARE AVAILABLE WITH ANISN.

156 EXTERMINATOR/EXTERMINATOR2 THE MULTIGROUP, TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED IN X-Y, R-Z, OR R-THETA GEOMETRY.

161 FORTRAN TDC TDC SOLVES THE BOLTZMANN EQUATION IN MULTIGROUP FORM FOR THE TRANSPORT OF NEUTRONS OR THE ADJOINT EQUATION IN FINITE (R,Z) CYLINDRICAL GEOMETRY BY THE DISCRETE SN METHOD. THE PROBLEM MAY BE HOMOGENEOUS (NO SOURCES INDEPENDENT OF FLUXES) OR INHOMOGENEOUS, BUT ALL SOURCES MUST BE ISOTROPIC. NEUTRON SCATTERING MUST ALSO BE ISOTROPIC. A HOMOGENEOUS PROBLEM MAY BE SOLVED FOR THE EIGENVALUE K-EFF OR THE EIGENVALUE ALPHA (TIME CONSTANT). ALTERNATIVELY, THE HOMOGENEOUS PROBLEM MAY BE SOLVED FOR THE SIZE (RADIUS, HEIGHT OR BOTH) OF THE SYSTEM CORRESPONDING TO A SPECIFIED K-EFF OR FOR THE ATOM CONCENTRATION OF SOME MATERIAL CORRESPONDING TO A SPECIFIED EIGENVALUE K-EFF OR ALPHA. FOR INHOMOGENEOUS PROBLEMS, THE IMPOSED SOURCE MAY BE AN ISOTROPIC VOLUME-DISTRIBUTED SOURCE OR AN ISOTROPIC SHELL SOURCE ON THE OUTER BOUNDARY.

167 FLARE FLARE IS AN INEXPENSIVE CALCULATIONAL METHOD TO DETERMINE CORE REACTIVITY AND CORE POWER DISTRIBUTION. A SCOPING CALCULATION OF THIS TYPE IS VALUABLE IN APPRAISING THE PHYSICS CHARACTERISTICS OF PLANNED TEST MODES OF OPERATION SO THAT DETAILED ANALYSIS CAN BE RESERVED FOR THOSE CORE CALCULATIONS OF GREATER INTEREST FROM EITHER A TECHNICAL OR SAFETY STANDPOINT.

173 2DF 2DF IS A TWO-DIMENSIONAL MULTIGROUP PROGRAM WRITTEN IN FORTRAN FOR SOLVING THE NEUTRON TRANSPORT EQUATION USING THE SN METHOD. THE PROGRAM CAN DETERMINE THE REAL OR ADJOINT SOLUTION FOR X-Y, R-Z, OR R-THETA GEOMETRY. ISOTROPIC OR A FORM OF LINEAR ANISOTROPIC SCATTERING MAY BE CONSIDERED. VARIOUS BOUNDARY CONDITIONS ARE ALLOWED. THE PROGRAM ALSO CONTAINS A NUMBER OF SEARCH OPTIONS WHEREBY ONE CAN VARY DIMENSIONS OR CONCENTRATIONS TO ARRIVE AT A PREDETERMINED EIGENVALUE. A DISTRIBUTED SOURCE MAY BE SPECIFIED. A LIBRARY OF CROSS SECTIONS IS AVAILABLE ON MAGNETIC TAPE. CROSS SECTIONS MAY BE READ FROM THE LIBRARY TAPE AND/OR FROM CARDS.

192 2DXYL THE 2DXY PROGRAM (ACC ABSTRACT 18) HAS BEEN CONVERTED FROM FLOCO TO FORTRAN 63 FOR USE ON THE CDC1604 WITH CHANGES TO PERMIT THE INCLUSION OF FIXED SOURCE TERMS FROM TDC - TERMS REPRESENTING THE EFFECTIVE NET LOSS PER UNIT VOLUME DUE TO AXIAL LEAKAGE. IN THIS MANNER, A THREE-DIMENSIONAL FLUX SYNTHESIS CODE IS ACHIEVED.

199 TCP TDP IS A TWO-DIMENSIONAL LINEAR PERTURBATION THEORY CODE WHICH CALCULATES REACTIVITY COEFFICIENTS, PROMPT NEUTRON LIFETIMES, AND EFFECTIVE DELAYED FRACTIONS USING FLUXES FROM TDC CYLINDRICAL (R-Z) OR RECTANGULAR (X-Y) GEOMETRY.

209 DTF4 THE LINEAR, TIME-INDEPENDENT, BOLTZMANN EQUATION FOR PARTICLE TRANSPORT IS SOLVED FOR THE ENERGY, SPACE, AND ANGULAR DEPENDENCE OF THE PARTICLE DISTRIBUTION IN ONE-DIMENSIONAL SLABS, CYLINDERS, AND SPHERES. INDEPENDENT SOURCE OR EIGENVALUE (MULTIPLICATION, TIME-ABSORPTION, ELEMENT CONCENTRATION, ZONE THICKNESS OR SYSTEM DIMENSION) PROBLEMS ARE SOLVED SUBJECT TO VACUUM, REFLECTIVE, OR PERIODIC BOUNDARY CONDITIONS. A COMPLETE ENERGY-TRANSFER SCATTERING MATRIX IS ALLOWED FOR EACH LEGENDRE COMPONENT OF THE SCATTERING CROSS SECTION MATRICES.

211 MGDSN MGDSN IS A MODIFICATION OF DSN, THE ONE-DIMENSIONAL THEORY CODE DESIGNED TO ACCOMMODATE 100 GROUP ISOTROPIC MATERIAL CROSS SECTION DATA TAPES PREPARED BY CSP1 AND CSP2A.

212 VARI-QUIR THE TIME-DEPENDENT, MULTI-GROUP, TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED IN X-Y OR R-Z GEOMETRY.

220 GASP2 SOLVES FOR THE ONE-DIMENSIONAL DISTRIBUTION OF FISSILE AND FERTILE MATERIALS IN A NUCLEAR REACTOR WHICH WILL YIELD ANY DESIRED POWER DISTRIBUTION AND APPROXIMATELY RETAIN THIS DESIRED POWER DISTRIBUTION DURING THE BURN-UP HISTORY OF THE REACTOR CORE. A POISON AND POISON DISTRIBUTION SEARCH FOR A DESIRED MULTIPLICATION AND MINIMUM POWER DISTRIBUTION PERTURBATION CAN ALSO BE PERFORMED.

222 GAMBLE5/GAMBLE4 THE HOMOGENEOUS TWO-DIMENSIONAL MULTIGROUP DIFFUSION THEORY EQUATIONS WITH ARBITRARY GROUP-TO-GROUP SCATTERING AND ARBITRARY FISSION TRANSFER ARE SOLVED FOR HETEROGENEOUS ASSEMBLIES IN X-Y AND R-Z GEOMETRY. HOMOGENEOUS LOGARITHMIC BOUNDARY CONDITIONS ARE USED AT THE OUTER SURFACE OF THE ASSEMBLY AND AT THE SURFACE OF NON-DIFFUSION REGIONS. THE RESULTS INCLUDE THE GROUP AND POINT DEPENDENT NEUTRON FLUXES, THE POWER DISTRIBUTION, THE NEUTRON MULTIPLICATION FACTOR (K-EFFECTIVE), AND A DETAILED NEUTRON BALANCE.

241 HFN HFN SOLVES THE HOMOGENEOUS OR INHOMOGENEOUS ONE-DIMENSIONAL MULTIGROUP DIFFUSION EQUATION FOR ITS LOWEST EIGENVALUE AND THE CORRESPONDING DIRECT AND/OR ADJOINT EIGENVECTORS. INHOMOGENEOUS BOUNDARY CONDITIONS AND A FLEXIBLE SCATTER-TRANSFER MATRIX STRUCTURE ARE INCLUDED. OPTIONAL CALCULATIONS INCLUDE CRITICALITY SEARCHES, DETECTOR ACTIVATION TRAVERSES, AND INTEGRALS FOR PERTURBATION THEORY ANALYSIS.

262 MACH1 MACH1 PERFORMS ONE-DIMENSIONAL MULTIGROUP DIFFUSION SOLUTIONS AND ASSOCIATED CALCULATIONS, INCLUDING CRITICALITY SEARCHES, PERTURBATION, REACTION SUMMARY, BETA EFFECTIVE, GROUP COLLAPSING, AND POINTWISE REACTION RATES AND RATIOS. SEVERAL CARD DUMPS OF COMPUTED DATA ARE AVAILABLE ON OPTION.

264 VARI-QUIR3 THE STEADY-STATE, MULTIGROUP, TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED IN X-Y, R-Z, AND R-THETA GEOMETRY.

270 CAESAR4/LIBLST CAESAR4 SOLVES THE ONE-DIMENSIONAL, MULTIGROUP DIFFUSION EQUATIONS IN ANY OF THREE GEOMETRIES AND PROVIDES A WIDE CHOICE OF BOUNDARY CONDITIONS, CRITICALITY SEARCHES, EDITS AND OTHER AUXILIARY COMPUTATIONS.

287 BISYN BISYN SOLVES THE TWO-DIMENSIONAL MULTIGROUP NEUTRON DIFFUSION EQUATIONS IN X-Y OR R-Z GEOMETRY USING A NONITERATIVE SYNTHESIS METHOD. THIS APPROACH IS DESIGNED TO GREATLY REDUCE THE COMPUTER COST OF RUNNING TWO-DIMENSIONAL MULTIGROUP PROBLEMS AT THE RISK OF SOME LOSS IN ACCURACY OF THE DETAILED FLUX DISTRIBUTION.

288 SNARG-1D THE PROGRAM IS DESIGNED TO SOLVE THE ONE-DIMENSIONAL NEUTRON TRANSPORT EQUATIONS. SNARG-1D IS WRITTEN FOR THE SOLUTION OF ONE-DIMENSIONAL PROBLEMS USING THE ORDER $N = 2, 4, 6, 8, 12, 16, \text{ OR } 32$, AND APPLICABLE TO PLANE, CYLINDRICAL AND SPHERICAL GEOMETRIES. THE REAL OR ADJOINT SOLUTION MAY BE CALCULATED AND HOMOGENEOUS OR INHOMOGENEOUS PROBLEMS MAY BE SOLVED. FOUR CRITICALITY SEARCH OPTIONS ARE PROVIDED FOR WHICH A FIXED K_{EFF} OR ALPHA, INVERSE PERIOD, VALUE MAY BE SPECIFIED RATHER THAN THE CRITICALITY VALUE, $K_{\text{EFF}} = 1$. EITHER OF TWO INHOMOGENEOUS SOLUTIONS MAY BE OBTAINED (SHELL OR DISTRIBUTED SOURCE CALCULATIONS). LINEAR ANISOTROPIC COMPONENTS OF BOTH THE SHELL SOURCE AND THE SCATTERING CROSS SECTIONS MAY BE INCLUDED. FISSION SPECTRUM MATRICES ARE ALLOWED AS BOTH MATERIAL-DEPENDENT AND INCIDENT NEUTRON ENERGY-DEPENDENT FUNCTIONS.

304 PERT4 PERT4 COMPUTES REACTIVITY COEFFICIENT TRAVERSES IN X-Y, R-Z, OR R-THETA GEOMETRY USING THE FIRST-ORDER PERTURBATION EQUATIONS IN THE DIFFUSION APPROXIMATION. FLUX AND ADJOINT INPUT CAN BE TAKEN DIRECTLY FROM 2-D CALCULATIONS OR SYNTHESIZED FROM RADIAL AND AXIAL 1-D CALCULATIONS. THE CODE CAN ALSO BE USED TO COMPUTE ACTIVITY TRAVERSES FOR ANY CROSS SECTION OF ANY MATERIAL, THE NEUTRON GENERATION TIME, AND THE EFFECTIVE DELAYED NEUTRON FRACTION.

312 TDSN THE LINEAR, TIME-INDEPENDENT, BOLTZMANN EQUATION IS SOLVED FOR THE ENERGY, SPACE, AND ANGULAR DEPENDENCE OF THE NEUTRON DISTRIBUTION IN ONE-DIMENSIONAL SLABS, CYLINDERS OR SPHERES OR IN TWO-DIMENSIONAL X-Y OR R-Z GEOMETRY. FIXED SOURCE OR MULTIPLICATION FACTOR (ADJOINT OR FLUX) PROBLEMS ARE SOLVED SUBJECT TO VACUUM, PLANE REFLECTIVE, ISOTROPIC REFLECTIVE OR 180 DEGREE ROTATIONALLY SYMMETRIC BOUNDARY CONDITIONS. A COMPLETE ENERGY TRANSFER SCATTERING MATRIX IS ALLOWED FOR EACH LEGENDRE COMPONENT OF SCATTERING CROSS SECTION MATRICES THROUGH P1.

319 GASP7 GASP7 CALCULATES THE ONE-DIMENSIONAL DISTRIBUTION OF FISSILE AND FERTILE MATERIALS IN A NUCLEAR REACTOR WHICH WILL YIELD ANY DESIRED POWER DISTRIBUTION DURING THE BURNUP HISTORY OF THE REACTOR CORE. A POISON AND POISON DISTRIBUTION SEARCH FOR A DESIRED MULTIPLICATION AND MINIMUM POWER DISTRIBUTION PERTURBATION CAN ALSO BE PERFORMED.

320 TEMCO7 TEMCO7 COMPUTES REACTOR TEMPERATURE COEFFICIENTS.

D. DEPLETION, FUEL MANAGEMENT, COST ANALYSIS, AND REACTOR ECONOMICS

6/68

55 AIMFIRE THIS PROGRAM WAS DESIGNED TO COMPARE THE COSTS OF VARIOUS FUEL CYCLES. THE PROGRAM CONTAINS A LIBRARY OF FAST AND THERMAL MICROSCOPIC CROSS SECTIONS, DECAY CONSTANTS, AND FISSION YIELDS FOR 50 ISOTOPES. THE PRESENT VERSION IS USED TO INVESTIGATE THE ECONOMICS OF URANIUM FUEL SYSTEMS.

58 SIZZLE SIZZLE SOLVES THE ONE-DIMENSIONAL, OR MULTIGROUP BURNUP PROBLEM IN THE DIFFUSION THEORY APPROXIMATION FOR FAST INTERMEDIATE REACTORS. AFTER THE INITIAL CALCULATION AT T=0, AVERAGE CROSS SECTIONS ARE COMPUTED FOR FURTHER CALCULATIONS USING ONE TO SIX ENERGY GROUPS. CRITICALITY MAY BE MAINTAINED BY USE OF A CONCENTRATION SEARCH. THE CONCENTRATION OF THE VARIOUS ISOTOPES IS PERMITTED TO VARY ONLY FROM REGION-TO-REGION. CHAINS INCLUDED ARE TH232, U238, AND A FISSION PRODUCT POISON CHAIN.

99 DUO DIMENSIONAL BURNOUT(DDB) THE FIVE-GROUP, TWO-DIMENSIONAL, NEUTRON DIFFUSION EQUATIONS IN CYLINDRICAL GEOMETRY ARE SOLVED WITH BURNOUT OPTIONS AND CONTROL ROD SEARCH OPTIONS.

117 FEVER FEVER PERFORMS ONE-DIMENSIONAL FEW-GROUP DEPLETION CALCULATIONS. OPTIONS ARE AVAILABLE TO ADJUST CONTROL POISONS IN VARIOUS REGIONS OF THE REACTOR, SELF-SHIELDING OF LUMPED POISONS, AND TO CALCULATE HOT MAXIMUM AND COLD SHUT-DOWN MULTIPLICATION.

134 NUCY THE CALCULATION OF NUCLIDE
CONCENTRATIONS AT A POINT IN A REACTOR AT SUCCESSIVE TIME INTER-
VALS, WITH EXPOSURE TO A TWO-GROUP NEUTRON FLUX. INFINITE SYSTEM
CRITICALITY IS CALCULATED.

146 NPRFCCP NUCLEAR FUEL CYCLE COSTS IN DOLLARS PER YEAR (\$/YR), AND IN MILLS PER KILOWATT-HOUR (MILLS/KWHR) ARE COMPUTED AND TABULATED FOR EACH REGION OF A MULTIREGION REACTOR CORE, ON THE BASIS OF AEC-LEASED NUCLEAR FUEL MATERIAL AND OF PRIVATELY-OWNED NUCLEAR FUEL MATERIAL. FUEL CYCLE COSTS ARE COMPUTED SEPARATELY FOR EACH REGION OR ZONE, FOR CORE DESIGNS OF ANY CONFIGURATION OR COMBINATION OF MATERIAL DEPLETION OR ENRICHED URANIUM FUEL OR OTHER SPECIAL NUCLEAR MATERIAL. PRINTED OUTPUT OF THE PROGRAM INCLUDES (A) DETAILED FUEL CYCLE COSTS FOR EACH ZONE IN TABULAR FORM, FOR AEC-LEASED AND PRIVATELY-OWNED NUCLEAR FUEL MATERIAL, RESPECTIVELY, (B) A SUMMARY TABULATION OF NUCLEAR FUEL COSTS FOR ALL REGIONS OF THE COMPLETE CORE, INCLUDING FIXED CHARGES ON WORKING CAPITAL REQUIRED FOR CORE FABRICATION AND FOR NUCLEAR FUEL MATERIAL, (C) A SUMMARY TABULATION OF CERTAIN COMPUTED PERFORMANCE AND ECONOMIC DATA, VIZ., AVERAGE RESIDENCE TIME, ANNUAL FUEL THROUGHPUT, UNIT ELECTRICAL ENERGY YIELD, ANNUAL POWER GENERATION, AND CORE FABRICATION COSTS, AND (D) A TABULATION OF ALL INPUT DATA FOR ALL REGIONS. THE PRINTING OF DATA DESCRIBED IN (A) AND (D) ABOVE IS OPTIONAL WITH PROGRAM USE.

179 ISCTOPES THIS PROGRAM CAN BE USED TO CALCULATE FOR ANY NEUTRON FLUX THE OPTIMUM TIME OF IRRADIATION FOR MAXIMUM YIELD, THE SPECIFIC ACTIVITY OF THE PRODUCT ISOTOPE IN CURIES PER GRAM OF TARGET MATERIAL, AND THE COMBINED SPECIFIC ACTIVITY OF THE TARGET AND PRODUCT ISOTOPES. THE PRODUCT ISOTOPE MAY BE PRODUCED BY ANY SIMPLE REACTION SUCH AS (N,GAMMA), (N,P), (N,2N), ETC., OR IT MAY BE PRODUCED BY DECAY OF A PARENT ISOTOPE.

180 ISOCRUNCH ISOCRUNCH CAN BE USED TO COMPUTE THE AMOUNT OF EACH ISOTOPE IN A REACTION AND DECAY CHAIN FOR ANY SPECIFIED NEUTRON FLUX AND TIME, TO SUM THE CONTRIBUTIONS OF VARIOUS CHAINS TO THE SAME ISOTOPE, TO GRAPH ON AN ASSOCIATED ELECTROPLOTTER OR CALCOMP THE YIELD OF AN ISOTOPE VS. TIME FOR A GIVEN FLUX, AND TO FIND THE OPTIMUM TIME FOR MAXIMUM YIELD OF AN ISOTOPE IN A CHAIN. THE PROGRAM DOES NOT TAKE INTO ACCOUNT THE SELF-SHIELDING OF A TARGET IN A REACTOR OR THE DEPENDENCE OF REACTION CROSS SECTIONS ON NEUTRON ENERGY WHICH CAN BE HANDLED BY ADJUSTING THE INPUT DATA.

221 RELOAD-FEVER A FEW-GROUP, 1-D DEPLETION CALCULATION WHICH ALLOWS FUEL IN VARIOUS STAGES OF IRRADIATION TO BE HOMOGENIZED INTO THE SAME REGION FOR PURPOSES OF THE DIFFUSION CALCULATION BUT FOLLOWS THE DEPLETION OF EACH OF THE SUB-REGIONS SEPARATELY. THE CALCULATION MAY BE INTERRUPTED PERIODICALLY FOR REFUELING ONE OR MORE REGIONS. RECYCLING IS OPTIONAL AND THERE IS NO LIMIT TO THE NUMBER OF REFUELINGS WHICH MAY BE PERFORMED. A CONTROL POISON SEARCH IS AVAILABLE AND CONCENTRATION DEPENDENT SELF-SHIELDING FACTORS MAY BE APPLIED TO ONE LUMPED POISON.

223 REVISED-GAD THIS INFINITE-MEDIUM DEPLETION PROGRAM PERFORMS FUEL CYCLE CALCULATIONS ON REACTORS EMPLOYING PARTIAL REFUELING. THE BURN-UP OF UP TO 12 DISCRETE FUEL COMPOSITIONS (REGIONS OR STAGES OF IRRADIATION) MAY BE FOLLOWED SIMULTANEOUSLY. THE BURN-UP CALCULATION MAY BE INTERRUPTED PERIODICALLY TO REMOVE THE CONTENTS AND TO REFUEL ONE OR MORE REGIONS. FUEL MAY BE PARTIALLY OR COMPLETELY RECYCLED.

224 WAMPUM THIS PROGRAM CALCULATES FUEL CYCLE COSTS ON A DETAILED BASIS, USING RESULTS OF NUCLEAR DEPLETION CALCULATIONS AND CERTAIN SPECIFIED ECONOMICS ASSUMPTIONS. THE PURPOSE IS TO PROVIDE A MEASURE OF PERFORMANCE FOR COMPARING OR OPTIMIZING FUEL CYCLES AND ASSOCIATED REACTOR CORE AND FUEL ELEMENT CHARACTERISTICS.

225 TEMCO TEMCO COMPUTES REACTOR TEMPERATURE COEFFICIENTS.

226 OPUS THE CODE GENERATES A FLOW NETWORK EQUIVALENT TO A GAS-COOLED NUCLEAR POWER PLANT OF SPECIFIED ELECTRICAL OUTPUT (IN THE RANGE OF 100 TO 1000 MW) ACCORDING TO INPUT DATA AND PROGRAMMED RULES, PROCEEDS TO EVALUATE THE PLANT PERFORMANCE AND PRICE OF THE TURBOGENERATOR SET (ACCORDING TO GENERAL ELECTRIC PRICE DATA), AND PRINTS AS A RESULT A CODED LIST OF ALL PLANT COMPONENTS AND A DETAILED PERFORMANCE MAP.

227 STMGEN STMGEN WILL BE USED IN A PLANT OPTIMIZATION PROGRAM. THIS CODE DETERMINES THE AREA OF EACH SECTION OF A STEAM GENERATOR REQUIRED TO SATISFY THE DESIGN CONDITIONS OF HEAT TRANSFER, PRESSURE DROP AND MAXIMUM TUBE TEMPERATURE CONSTRAINTS. THE COST OF THE GENERATOR IS COMPUTED AS A FUNCTION OF THE TOTAL HEAT TRANSFER AREA, THE NON-PRODUCTIVE TUBE LENGTH REQUIRED TO CONNECT THE HEADERS, PLUS THE COST OF THE HEADERS.

231 RAD2 THIS PROGRAM CALCULATES THE FISSION PRODUCT ACTIVITY DISTRIBUTIONS IN A HIGH TEMPERATURE GAS-COOLED REACTOR SYSTEM.

240 ASSAULT ASSAULT IS A MULTIGROUP, TWO-DIMENSIONAL REACTOR DEPLETION PROGRAM. GIVEN NUCLIDE CONCENTRATIONS AND MICROSCOPIC CROSS SECTIONS, THE STEADY-STATE MULTI-REGION, MULTIGROUP DIFFUSION EQUATIONS ARE SOLVED IN ONE OR TWO DIMENSIONS OVER A FINITE-DIFFERENCE MESH POINT SYSTEM. CALCULATED NEUTRON FLUXES ARE THEN USED TO DETERMINE NUCLIDE CONCENTRATIONS AFTER A SPECIFIED PERIOD OF EXPOSURE. THESE CALCULATIONS ARE REPEATED FOR A SPECIFIED NUMBER OF TIME-STEPS.

260 GARGOYLE GARGOYLE IS AN INFINITE MEDIUM DEPLETION CODE FOR FUEL CYCLE CALCULATIONS ON REACTORS EMPLOYING PARTIAL REFUELING. THE BURNUP OF UP TO 12 DISCRETE FUEL COMPOSITIONS (REGIONS OR STAGES OF IRRADIATION) MAY BE FOLLOWED SIMULTANEOUSLY. THE BURNUP CALCULATION MAY BE INTERRUPTED PERIODICALLY TO REMOVE THE CONTENTS OF AND TO REFUEL ONE OR MORE REGIONS. FUEL MAY BE COMPLETELY OR PARTIALLY RECYCLED. A CONTROL POISON SEARCH MAY BE PERFORMED AT EACH TIME-STEP. FEED FUEL SEARCHES ARE PERMITTED AT THE END OF EACH BURNUP CYCLE BEFORE REFUELING. CONCENTRATION DEPENDENT SELF-SHIELDING FACTORS MAY BE APPLIED TO ANY NUCLIDE EXCEPT MODERATORS. THE NUCLIDE SCHEME EMPLOYED, ALTHOUGH NOT COMPLETELY GENERAL, IS FLEXIBLE.

269 DTF-BURN DTF-BURN IS A ONE-DIMENSIONAL, MULTIGROUP BURNUP CODE BASED ON TRANSPORT THEORY. A DIFFERENT TYPE OF EIGENVALUE CALCULATION (MULTIPLICATION, PERIOD, NUCLIDE CONCENTRATION, ZONE THICKNESS, OR SYSTEM DIMENSION) CAN BE PERFORMED AT THE BEGINNING OF EACH OF A SPECIFIED SET OF TIME-STEPS. THIS FEATURE PERMITS THE SIMULATION, IN A SINGLE PROBLEM RUN, OF A SYSTEM IN WHICH VARIOUS CONTROL METHODS ARE USED AS A FUNCTION OF TIME. THE CONCENTRATION OF THE VARIOUS NUCLIDES IS PERMITTED TO VARY FROM REGION-TO-REGION OR POINT-BY-POINT. NUCLIDES CONSIDERED ARE U235, U236, U238, NP239, PU239, PU240, PU241, PU242, AND FP, A PSEUDO FISSION PRODUCT PAIR.

R275 PDQ7 PDQ7 SOLVES FEW-GROUP NEUTRON DIFFUSION-DEPLETION PROBLEMS IN ONE, TWO, AND THREE DIMENSIONS. ADJOINT SOLUTIONS ARE ALSO AVAILABLE AND TWO OVERLAPPING THERMAL GROUPS MAY BE USED IN ONE AND TWO-DIMENSIONAL PROBLEMS. EITHER POINTWISE OR REGIONWISE DEPLETION MAY BE PERFORMED USING THE HARMONY DEPLETION SYSTEM. THE GEOMETRY MAY BE RECTANGULAR, CYLINDRICAL, OR SPHERICAL IN ONE DIMENSION, RECTANGULAR, CYLINDRICAL, OR HEXAGONAL IN TWO DIMENSIONS, AND RECTANGULAR OR HEXAGONAL IN THREE DIMENSIONS. ALL GEOMETRIES PROVIDE FOR VARIABLE MESH SPACING IN ALL DIMENSIONS. ZERO FLUX, ZERO CURRENT, AND ROTATIONAL SYMMETRY BOUNDARY CONDITIONS ARE AVAILABLE, AND BOUNDARY VALUE PROBLEMS MAY BE SOLVED BY SPECIFYING THE FLUX VALUES ON ONE OR MORE BOUNDARIES.

301 FREVAP6 THE FREVAP TYPE OF CODE FOR ESTIMATING THE RELEASE OF LONGER-LIVED METALLIC FISSION PRODUCTS FROM HTGR FUEL ELEMENTS HAS BEEN DEVELOPED TO TAKE INTO ACCOUNT THE COMBINED EFFECTS OF THE RETENTION OF METALLIC FISSION PRODUCTS BY FUEL PARTICLES AND THE RATHER STRONG ABSORPTION OF THESE FISSION PRODUCTS BY THE GRAPHITE OF THE FUEL ELEMENTS. RELEASE CALCULATIONS ARE MADE ON THE BASIS THAT THE LOSS OF FISSION PRODUCT NUCLIDES SUCH AS OF STRONTIUM, CESIUM, AND BARIUM IS DETERMINED BY THEIR EVAPORATION FROM THE GRAPHITE SURFACES AND THEIR TRANSPIRATION INDUCED BY THE FLOWING HELIUM COOLANT. THE CODE IS DEvised SO THAT CHANGES OF FISSION RATE (FUEL ELEMENT POWER), FUEL TEMPERATURE, AND GRAPHITE TEMPERATURE MAY BE INCORPORATED INTO THE CALCULATION. TEMPERATURE IS QUITE IMPORTANT IN DETERMINING RELEASE BECAUSE, IN GENERAL, BOTH RELEASE FROM FUEL PARTICLES AND LOSS BY EVAPORATION (TRANSPIRATION) VARY EXPONENTIALLY WITH THE RECIPROCAL OF THE ABSOLUTE TEMPERATURE.

302 GAFFE A ZERO-DIMENSIONAL CALCULATION OF FEED FUEL REQUIREMENTS IS PERFORMED TO PRODUCE A SPECIFIED END OF CYCLE MULTIPLICATION FACTOR FOR THE EQUILIBRIUM FUEL CYCLE, GIVEN FEED COMPOSITION, LENGTH OF CYCLE AND REACTOR POWER. IT IS ALTERNATELY POSSIBLE TO COMPUTE CYCLE LENGTH OR FEED ENRICHMENT. THE CODE IS A SURVEY TOOL WHICH ASSUMES PERIODIC REFUELING AND PERMITS COMPLETE OR PARTIAL RECYCLING OF MATERIALS. THE SEGREGATED FUEL CONCEPT CAN BE HANDLED WITHIN THE FRAMEWORK OF THE CALCULATION.

313 CINDER (M0102) CINDER IS A NINE-GROUP, ONE-POINT DEPLETION AND FISSION PRODUCT PROGRAM BASED ON THE EVALUATION OF A GENERAL ANALYTICAL SOLUTION OF NUCLIDES COUPLED IN ANY LINEAR SEQUENCE OF RADIOACTIVE DECAYS AND NEUTRON ABSORPTIONS IN A SPECIFIED NEUTRON FLUX SPECTRUM. THE DESIRED DEPLETION AND FISSION PRODUCT CHAINS ARE ALL PHYSICAL DATA AND ARE SPECIFIED BY THE PROBLEM ORIGINATOR. THE PROGRAM COMPUTES INDIVIDUAL NUCLIDE NUMBER DENSITIES, ACTIVITIES, DISINTEGRATION RATES, AND MACROSCOPIC AND BARNS/FISSION POISONS AT EACH TIME-STEP AS WELL AS SELECTED SUMMARIES OF THESE DATA.

314 NAP NAP CALCULATES THE SPECTRUM AND SPATIAL DISTRIBUTION IN ONE DIMENSION OF ACTIVATION GAMMA RAYS FOLLOWING NEUTRON IRRADIATION.

318 FEVER7 FEVER7 PERFORMS A MULTIGROUP, 1-DIMENSIONAL DEPLETION CALCULATION WHICH ALLOWS FUEL IN VARIOUS STAGES OF IRRADIATION TO BE HOMOGENIZED INTO THE SAME REGION FOR PURPOSES OF THE DIFFUSION CALCULATION BUT FOLLOWS THE DEPLETION OF EACH OF THE SUBREGIONS SEPARATELY. THE CALCULATION MAY BE INTERRUPTED PERIODICALLY FOR REFUELING ONE OR MORE REGIONS. RECYCLING IS OPTIONAL AND THERE IS NO LIMIT TO THE NUMBER OF REFUELINGS WHICH MAY BE PERFORMED. A CONTROL POISON SEARCH IS AVAILABLE AND CONCENTRATION DEPENDENT SELF-SHIELDING FACTORS MAY BE APPLIED TO A NUMBER OF LUMPED POISONS.

322 ISCSEARCH THIS PROGRAM WAS DEVELOPED TO CALCULATE THE UNKNOWN REACTION CROSS SECTION, FLUX VALUE, OR PRODUCT ACTIVITY IN AN ISOTOPE-PRODUCTION SCHEME CONSISTING OF TWO OR THREE NUCLIDES.

325 2DB 2DB IS A FLEXIBLE, TWO-DIMENSIONAL (X-Y, R-Z, R-THETA GEOMETRY) DIFFUSION CODE FOR USE IN FAST REACTOR ANALYSES. THE CODE CAN BE USED TO -

- (A) COMPUTE FUEL BURNUP,
- (B) PERFORM CRITICALITY SEARCHES ON TIME ABSORPTION (ALPHA), MATERIAL CONCENTRATIONS, AND REGION DIMENSIONS USING A REGULAR OR ADJOINT MODEL. CRITICALITY SEARCHES CAN BE PERFORMED DURING BURNUP TO COMPENSATE FOR FUEL DEPLETION.

R336 PCQ5 THE FEW-GROUP TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED. UP TO FIVE GROUPS MAY BE USED WITH SCATTERING ALLOWED BETWEEN ADJACENT GROUPS. IN ADDITION, DEPLETION PROBLEMS MAY BE SOLVED WITH PDQ5.

E. SPACE-INDEPENDENT KINETICS

6/68

121 AIREK3 AIREK3 FINDS THE NUMERICAL SOLUTION TO THE SPACE INDEPENDENT REACTOR KINETICS EQUATIONS BASED ON THE METHOD DEVELOPED BY E. R. COHEN. INPUT AND OUTPUT ARE SIMPLIFIED AND THE POWER, INVERSE PERIOD, FEEDBACKS, AND PRE-CURSORS ARE DISPLAYED GRAPHICALLY.

122 SNAPKIN5/SNAPKIN5A SNAPKIN5 PROVIDES A ONE-REGION TIME-DEPENDENT CALCULATION OF POWER, ENERGY, TEMPERATURE, REACTIVITY, INVERSE PERIOD, AND HYDROGEN LOSS IN A SNAP REACTOR AFTER A PERTURBATION FROM GIVEN INITIAL CONDITIONS. SNAPKIN-5A, IN ADDITION, WEIGHS POWER, HEAT CAPACITY, AND REACTIVITY IMPORTANCE FOR TWENTY-FIVE OR FEWER REGIONS.

135 TRAFICORPORATION EXPERIMENTAL FREQUENCY RESPONSE DATA OBTAINED FROM A LINEAR DYNAMIC SYSTEM IS PROCESSED TO OBTAIN THE TRANSFER FUNCTION AS A RATIO OF TWO FREQUENCY-DEPENDENT POLYNOMIALS. THE TRANSFER FUNCTION MAY HAVE NON-MINIMUM PHASE.

163 AIROS AIROS SOLVES THE SPACE-INDEPENDENT REACTOR KINETICS EQUATIONS AND PROVIDES FOR THE DETERMINATION OF REACTIVITY BY SOLVING IN ADDITION THE DISCRETIZED EQUATIONS WHICH REPRESENT THE SPATIAL HEAT AND MASS TRANSFER MODEL FOR SEVERAL FUEL CHANNELS. IN ADDITION, VARIATION OF THE FILM COEFFICIENT WITH FLOW IS ACCOUNTED FOR AS WELL AS THE PROVISION FOR FLOW DECAY AND AFTERGLOW HEATING. SCRAMS CAN BE INITIATED BY DELAYED SIGNALS FROM INSTRUMENTS WHICH SENSE ANY QUANTITY CALCULATED, E.G., POWER, INVERSE PERIOD OR TEMPERATURE. GENERALIZED FEEDBACK EQUATIONS ARE USED TO PROVIDE FLEXIBILITY IN THE MODELS THAT REPRESENT MULTICHANNEL HEAT TRANSFER INCLUDING CONDUCTION AND CONVECTION, ENERGY, PRESSURE AND OTHER PHENOMENON. THE REACTIVITY EQUATION IS ALSO GENERALIZED. THE REACTIVITY FEEDBACK COEFFICIENTS CAN BE CONSTANT OR VARY AS THE SQUARE ROOT OR RECIPROCAL OF TEMPERATURE. FURTHERMORE ANY FEEDBACK VARIABLE CAN BE USED TO INITIATE A REACTIVITY SCRAM, EACH WITH A UNIQUE DELAY TIME.

168 INVERSE KINETICS (R102) GIVEN THE SPACE-INDEPENDENT, ONE ENERGY GROUP REACTOR KINETICS EQUATIONS AND THE INITIAL CONDITIONS, THIS PROGRAM DETERMINES THE TIME VARIATION OF REACTIVITY REQUIRED TO PRODUCE THE GIVEN INPUT OF FLUX-TIME DATA.

188 CMPXMAT A SYSTEM OF N LINEAR EQUATIONS DERIVED FROM THE LAPLACE TRANSFORM OF A SET OF LINEARIZED DIFFERENTIAL EQUATIONS IS SOLVED FOR AMPLITUDE AND PHASE ANGLE AS A FUNCTION OF FREQUENCY.

255 R101 R101 SOLVES THE SPACE-INDEPENDENT, ONE-ENERGY GROUP REACTOR KINETICS EQUATIONS TO DETERMINE THE TIME VARIATION OF NEUTRON DENSITY GIVEN SPECIFIED INITIAL CONDITIONS. ANY OF FOUR PROGRAMMED REPRESENTATIONS OF EXCESS REACTIVITY CAN BE SELECTED.

290 GASA GASA DETERMINES THE STABILITY OF ANY PHYSICAL SYSTEM WHOSE MOTION IS DESCRIBABLE BY A SET OF FIRST ORDER LINEAR DIFFERENTIAL EQUATIONS.

303 BLCOST6 BLCOST6 COMBINES A REACTOR SPACE-INDEPENDENT KINETICS CODE WITH A TWO-DIMENSIONAL HEAT TRANSFER CODE, AND A TIME-DEPENDENT SPHERICAL GEOMETRY HEAT TRANSFER ROUTINE FOR FUEL PARTICLES. THE CODE IS APPLICABLE TO PROBLEMS FOR WHICH THE SPACE-INDEPENDENT FORM OF THE REACTOR KINETICS EQUATION IS APPLICABLE.

317 GAPOTKIN GAPOTKIN IS A POINT KINETICS CODE THAT SOLVES THE SPACE-INDEPENDENT KINETICS EQUATIONS FOR A VERY GENERAL FORM OF THE REACTIVITY FUNCTION.

326 AIROS2A AIROS2A SOLVES THE SPACE-INDEPENDENT REACTOR KINETICS EQUATIONS AND PROVIDES FOR THE DETERMINATION OF REACTIVITY BY SOLVING IN ADDITION THE DISCRETIZED EQUATIONS THAT REPRESENT THE SPATIAL HEAT AND MASS TRANSFER MODEL FOR SEVERAL FUEL CHANNELS. IN ADDITION, VARIATION OF THE FILM COEFFICIENT WITH FLOW IS ACCOUNTED FOR ALONG WITH THE PROVISION FOR FLOW DECAY AND AFTERGLOW HEATING. SCRAMS CAN BE INITIATED BY DELAYED SIGNALS FROM INSTRUMENTS THAT SENSE ANY QUANTITY CALCULATED, E.G., POWER, INVERSE PERIOD OR TEMPERATURE. GENERALIZED FEEDBACK EQUATIONS ARE USED TO PROVIDE FLEXIBILITY IN THE MODELS THAT REPRESENT MULTICHANNEL HEAT TRANSFER INCLUDING CONDUCTION AND CONVECTION, ENERGY, PRESSURE AND OTHER PHENOMENON SUCH AS FUEL MELTING, COOLANT BOILING AND VOIDING BURN-OUT. THE REACTIVITY EQUATION IS ALSO GENERALIZED. THE REACTIVITY FEEDBACK COEFFICIENTS CAN BE CONSTANT OR VARY AS THE SQUARE ROOT OR RECIPROCAL OF TEMPERATURE. FURTHERMORE, ANY FEEDBACK VARIABLE CAN BE USED TO INITIATE A REACTIVITY SCRAM, EACH WITH A UNIQUE DELAY TIME. AN INPUT GENERATOR COMPUTES THE CONDUCTION AND CONVECTION COEFFICIENTS FOR AN $N \times M$ NODAL, MULTICHANNEL SYSTEM USING BUILT-IN TABLES OF SPECIFIC HEAT, DENSITY, CONDUCTIVITY AND VISCOSITY FOR THE COMMON FUEL, STRUCTURE AND COOLANT MATERIALS, AND PERFORMS AN INITIAL TEMPERATURE CALCULATION. THE FILM COEFFICIENTS MAY BE SPECIFIED OR CALCULATED USING LYONS EQUATION OR THE DITTUS-BOELTER EQUATION.

F. SPACE-TIME KINETICS, COUPLED NEUTRONICS-HYDRODYNAMICS-THERMODYNAMICS, AND EXCURSION SIMULATIONS

6/68

102 AX1 AX1 PERFORMS A COUPLED NEU-
TRONICS-HYDRODYNAMICS CALCULATION. GIVEN A SPHERICALLY SYMMETRIC,
SUPERPRCMT CRITICAL SYSTEM, THE PROGRAM COMPUTES THE VARIATION IN
TIME AND SPACE OF THE SPECIFIC ENERGY, TEMPERATURE, PRESSURE,
DENSITY AND VELOCITY.

129 CONEC CONEC IS A COUPLED NEUTRONIC-ELASTICITY CODE DESIGNED FOR APPLICATION TO PULSED FAST REACTOR SYSTEMS. IT IS A ONE-DIMENSIONAL CALCULATION, CAPABLE OF DEALING WITH SOLID SPHERES OR SPHERICAL SHELLS. SPECIFICALLY, CONEC CALCULATES THE ALPHA OF A SYSTEM, THE TEMPERATURE DISTRIBUTION, THE RADIAL AND TANGENTIAL STRESS DISTRIBUTIONS, AND FROM THESE THE ACCELERATIONS, VELOCITIES, AND DISPLACEMENTS THROUGHOUT THE SYSTEM.

145 WEAK EXPLOSION PROGRAM THIS PROGRAM PERFORMS A COUPLED NEUTRONICS-HYDRODYNAMICS CALCULATION FOR A SPHERICALLY SYMMETRIC REACTOR CORE WITH A GIVEN COMPOSITION TO OBTAIN THE TIME-DEPENDENT ENERGY RELEASE THAT RESULTS FROM THE INSERTION OF REACTIVITY AT A GIVEN RATE. THE BASIC USE FOR THIS PROGRAM IS FOUND IN ANALYSIS OF FAST REACTOR CORE COLLAPSE ACCIDENTS.

153 HATCHET HATCHET IS A MAJOR MODIFICATION OF THE AX1 CODE DESIGNED TO STUDY BURST CHARACTERISTICS OF A SUPERPROMPT CRITICAL, CONCENTRIC SHELL, PULSED REACTOR. IT COMPUTES SPECIFIC ENERGY, TEMPERATURE, PRESSURE, DENSITY AND VELOCITY VARIATIONS AS A FUNCTION OF TIME AND SPACE. THE CODE ALSO COMPUTES REACTIVITY AS A FUNCTION OF INVERSE REACTOR PERIOD, POWER, THE TOTAL AND KINETIC ENERGIES, AND THE POSITION OF THE SHELLS WHICH COMPRISE THE SYSTEM.

174 FCRE/FORE2 FORE CALCULATES REACTOR POWER AND TEMPERATURES OF FUEL, COOLANT, CLAD, AND STRUCTURE AS FUNCTIONS OF TIME IN RESPONSE TO A PROGRAMMED REACTIVITY INSERTION SPECIFIED AS A SERIES OF RAMPS. TEMPERATURE PROFILES ARE COMPUTED AT SPECIFIED AXIAL POSITIONS FOR AN AVERAGE CHANNEL AND FOR THE PEAK POWER (CENTRAL) CHANNEL. THE HEAT OF FUSION ACCOMPANYING FUEL MELTING IS TAKEN INTO ACCOUNT. FEEDBACK REACTIVITY MECHANISMS THAT RESPOND TO CHANGES OF TEMPERATURES INCLUDE THE FUEL DOPPLER EFFECT AND THERMAL EXPANSION OF THE CORE (AND AXIAL BLANKET OR REFLECTOR) MATERIALS. FORE2 IS A COUPLED THERMAL HYDRAULICS-POINT KINETICS DIGITAL COMPUTER CODE DESIGNED TO CALCULATE SIGNIFICANT REACTOR PARAMETERS UNDER STEADY-STATE CONDITIONS, OR AS FUNCTIONS OF TIME DURING TRANSIENTS. THE TRANSIENTS MAY RESULT FROM A PROGRAMMED REACTIVITY INSERTION OR A POWER CHANGE. VARIABLE INLET COOLANT FLOW RATE AND TEMPERATURE ARE CONSIDERED. THE CODE CALCULATES THE REACTOR POWER, THE INDIVIDUAL REACTIVITY FEEDBACKS, AND THE TEMPERATURE OF COOLANT, CLADDING, FUEL, STRUCTURE, AND ADDITIONAL MATERIAL FOR UP TO SEVEN AXIAL POSITIONS IN THREE CHANNEL TYPES WHICH REPRESENT RADIAL ZONES OF THE REACTOR. THE HEAT OF FUSION ACCOMPANYING FUEL MELTING, THE LIQUID METAL VOIDING REACTIVITY, AND THE SPATIAL AND THE TIME VARIATION OF THE FUEL CLADDING GAP COEFFICIENT DUE TO CHANGES IN GAP SIZE ARE CONSIDERED.

187 PUMP IMPELLER DESIGN SYSTEM THIS SYSTEM HAS BEEN DEVELOPED FOR THE DESIGN AND DEVELOPMENT OF HIGH SPEED TURBOMACHINERY FOR PUMPING HIGH TEMPERATURE LIQUID METALS. AFTER INITIALING THE ANALYSIS AND DESIGN EFFORT BY ESTABLISHING THE PUMP CONDITIONS OF SERVICE THE CODES ARE USED TO DEFINE THE PUMP IMPELLER DESIGN. IN ALL PROGRAMS, REAL FLUID EFFECTS ARE NOT CONSIDERED. THE MACHINING CODE PRODUCES A PAPER TAPE FOR USE IN THE ACTUAL FABRICATION OF THE IMPELLER.

191 AX-TNT

AX-TNT SOLVES

- (A) THE COUPLED HYDRODYNAMIC, THERMODYNAMIC AND NEUTRONIC EQUATIONS WHICH DESCRIBE A SPHERICAL, SUPER-PROMPT CRITICAL REACTOR SYSTEM DURING AN EXCURSION,
- (B) THE COUPLED EQUATIONS OF MOTION, AND IDEAL GAS EQUATION OF STATE FOR THE DETONATION OF A SPHERICAL CHARGE IN A GAS.

R274 WIGL2

WIGL2 IS A ONE-DIMENSIONAL TWO-GROUP SPACE-TIME DIFFUSION THEORY PROGRAM WITH ZERO, ONE, OR SIX DELAYED NEUTRON GROUPS. THE PROGRAM WILL TREAT SLAB, CYLINDRICAL, AND SPHERICAL GEOMETRIES AND INCLUDES NON-BOILING HEAT TRANSFER. IT ACCOUNTS FOR XENON FEEDBACK AND FEEDBACK EFFECTS DUE TO FUEL AND COOLANT TEMPERATURE. CONTROL ROD MOTION AND CONTROL SYSTEM FEEDBACK BASED ON TOTAL CORE POWER OR OUTLET COOLANT TEMPERATURE CAN BE SIMULATED. TRANSIENTS MAY BE EXCITED BY PRESCRIBED CHANGES IN INLET COOLANT TEMPERATURE, COOLANT FLOW RATE, OR ROD POSITION.

293 MARS MARS PERFORMS A COUPLED NEUTRONICS-HYDRODYNAMICS CALCULATION FOR A FINITE CYLINDER CORE WITH CONCENTRIC REGIONS OF DIFFERENT COMPOSITIONS AND CHARACTERISTICS TO OBTAIN THE TIME-DEPENDENT ENERGY RELEASE THAT RESULTS FROM THE INSERTION OF REACTIVITY ACCORDING TO A PRESCRIBED PROGRAM. THE BASIC USE FOR THIS PROGRAM IS FOUND IN ANALYSIS OF FAST REACTOR CORE COLLAPSE ACCIDENTS. THE REACTIVITY CHANGES DUE TO MATERIAL DISPLACEMENT IS DETERMINED BY THE USE OF PERTURBATION THEORY AND THE PRESSURE GENERATION FOR THE HYDRODYNAMICS CALCULATION IS DETERMINED BY THE USE OF AN EXPONENTIAL FORM FOR THE SATURATED VAPOR PRESSURE CURVE. THE DOPPLER EFFECT IS DETERMINED BY THE USE OF A FLEXIBLE $T^{*}(-N)$ FORM WHERE N IS A PARAMETER THAT IS SPECIFIED AT THE TIME OF EXECUTION. THE EXTERNAL REACTIVITY INSERTION CAN BE IN THE FORM OF A STEP, A LINEAR RAMP OR A PARABOLIC RAMP, WITH A LIMIT ON THE AMOUNT OF REACTIVITY WHICH CAN BE INSERTED BEING A DEFINABLE QUANTITY. TABULATED VALUES OF THE POWER AND MATERIAL WORTH DISTRIBUTIONS ARE ACCEPTED AS INPUT. THE CHARACTERISTICS OF EACH REGION (DENSITY, SPECIFIC HEAT, DOPPLER COEFFICIENT, EQUATION OF STATE, GEOMETRIC LIMITS) ARE SPECIFIED INDEPENDENTLY FOR EACH REGION TO PROVIDE FLEXIBILITY IN THE TREATMENT OF CORES OF UNUSUAL CONFIGURATION. THE NEUTRON KINETICS CALCULATIONS ARE DONE WITH A POINT REACTOR MODEL AND DELAYED NEUTRONS ARE USED.

309 TSN THE TIME-DEPENDENT NEUTRON TRANSPORT EQUATION IS SOLVED. ENERGY DEPOSITION IS ALLOWED TO CAUSE VARIATION IN THE NEUTRON CROSS SECTIONS FOR THE CORE REGION, IN THE CORE DENSITY, AND IN CORE HEIGHT. THE RESULT IS A KINETICS CALCULATION INCLUDING SPATIAL DEPENDENCE BOTH IN FEEDBACK EFFECTS AND IN NEUTRON DENSITY. THE RESULTS ARE SUMMARIZED IN A MANNER SIMILAR TO POINT-KINETICS CODES, AND SPATIAL DISTRIBUTIONS ARE ALSO GIVEN. GRAPHICAL SUMMARIES OF THE SIGNIFICANT VARIABLES AND SPATIAL DISTRIBUTIONS ARE GIVEN. IN THE IBM360 VERSION, EITHER THE NEUTRON YIELD FROM FISSION OR THE THICKNESS OF A SPECIFIED ZONE CAN BE CHANGED AS AN INDEPENDENT FUNCTION OF TIME AS SPECIFIED BY THE USER. THIS PROVIDES TWO WAYS OF ALLOWING EXPLICIT REACTIVITY VARIATION WITH TIME.

310 GAKIN THE MULTIGROUP, 1-DIMENSIONAL, TIME-DEPENDENT DIFFUSION THEORY EQUATIONS ARE SOLVED IN SLAB, CYLINDRICAL OR SPHERICAL GEOMETRY WITH DELAYED NEUTRONS TAKEN INTO ACCOUNT. AN ARBITRARY SCATTERING MATRIX IS ALLOWED, TOGETHER WITH A PIECE-WISE, LINEAR, TIME-DEPENDENT, INHOMOGENEOUS SOURCE TERM. FEEDBACK IS AVAILABLE FROM TIME-DEPENDENT CROSS SECTION CHANGES AND BUILDUP IN XENON. THE TIME INTEGRATION IS DIVIDED INTO TIME ZONES WITH UNIQUE FEEDBACK AND SOURCE DATA FOR EACH ZONE.

G. RADIOLOGICAL SAFETY, HAZARD AND ACCIDENT ANALYSIS

6/68

47 CLCUD THE CLOUD PROGRAM CALCULATES THE EXTERNAL GAMMA-RAY DOSE RATE AND TOTAL INTEGRATED DOSE RESULTING FROM THE CONTINUOUS RELEASE OF RADIOACTIVE MATERIALS TO THE ATMOSPHERE.

172 AISITE2/R153 AISITE2 IS LARGELY BASED ON METHODS PROPOSED BY THE AEC IN TID-14844 BUT DIFFERS IN CERTAIN OF THE ASSUMPTIONS AND MODELS. THE CODE AUTOMATICALLY VARIES ANY ONE OF 46 PARAMETERS SUCH AS REACTOR POWER, BUILDING LEAK RATE, IODINE CLEAN-UP RATE, AND HALOGEN FILTER EFFICIENCY, COMPUTING THE EXCLUSION AREA, AND LOW POPULATION BOUNDARY ZONES AS FUNCTIONS OF THAT PARAMETER. THE EDIT INCLUDES DOSE VS. DISTANCE DATA, FRACTIONAL CONTRIBUTION BY ISOTOPE GROUP TO THE INHALATION DOSE, AND CRITICAL DISTANCES PROVIDING BOTH PRINTED AND GRAPHICAL DATA. THREE MODELS ARE AVAILABLE FOR FISSION PRODUCT RELEASE WITH UP TO 4 LEVELS OF CONTAINMENT.

196 CURIE/DOSE/THUNDERHEAD CURIE CALCULATES THE FISSION PRODUCT INVENTORY PRODUCED IN A REACTOR DURING CONSTANT POWER OPERATION AND RELEASES DIFFERENT PERCENTAGES OF THE NOBLE GASES, HALOGENS, AND PARTICULATES TO THE ATMOSPHERE AT REACTOR SHUTDOWN. DOSE CALCULATES THE TOTAL DOSE TO 13 INTERNAL BODY ORGANS RESULTING FROM INHALATION OF THE PASSING RADIOACTIVE CLOUD. THUNDERHEAD CALCULATES THE EXTERNAL CLOUD GAMMA EXPOSURE DOSE FROM THE RELEASED FISSION PRODUCTS.

200 SATURATED BLOWDOWN1 CALCULATION OF LOCAL PRESSURES, MASS FLOW RATES, FLUID QUALITIES, SPECIFIC VOLUMES, TEMPERATURES AND LOCAL HEAT TRANSFER COEFFICIENTS AT WETTED WALLS, DURING BLOWDOWN OF THE REACTOR PRIMARY COOLANT LOOP.

228 PRECON THE PURPOSE OF THE PROGRAM IS TO DETERMINE, FOR A GAS-COOLED REACTOR, THE CONTAINMENT PRESSURE AS A FUNCTION OF TIME, AFTER RUPTURES HAVE BEEN ASSUMED TO OCCUR IN THE PRIMARY SYSTEM OR IN CONJUNCTION WITH STEAM GENERATOR RUPTURES.

229 ELBOW DEFLECTIONS, RESTRAINT FORCES AND STRESSES ARE DETERMINED IN A FUEL ELEMENT SUBJECTED TO ASYMMETRIC POWER GENERATION, CIRCUMFERENTIALLY NON-UNIFORM COOLANT TEMPERATURE DISTRIBUTION, AND ASYMMETRIC FAST NEUTRON FLUX DAMAGE.

261 GADOSE/DOSET THE GADOSE PROGRAM CALCULATES RADIOACTIVITY AND DOSES RESULTING FROM INSTANTANEOUS ACCIDENTAL RELEASE OF ACTIVITY WHILE THE COMPANION PROGRAM DOSET INCLUDES THE EFFECTS OF A TIME-DEPENDENT ACCIDENTAL FISSION PRODUCT RELEASE FOR THE HTGR TYPE OF PLANT. GIVEN AN INITIAL FISSION PRODUCT INVENTORY RELEASED INTO A REACTOR VESSEL OR CONTAINMENT VOLUME, THE QUANTITY OF EACH ISOTOPE IS CALCULATED AT A NUMBER OF TIMES AT A NUMBER OF LOCATIONS (IN THE CONTAINMENT, CONTAINMENT RECIRCULATING CLEANUP FILTERS, LEAK COLLECTOR, PLANT EXHAUST FILTERS AND IN THE ATMOSPHERE AT A NUMBER OF DISTANCES FROM THE PLANT). EACH DECAY CHAIN IS CALCULATED SEPARATELY CONSIDERING BUILDUP AND DECAY OF EACH ISOTOPE. RADIOLOGICAL DOSES ARE CALCULATED FOR ANY THREE BODY ORGANS AND THE WHOLE BODY BASED ON METEOROLOGICAL AND PHYSIOLOGICAL INPUT PARAMETERS. FALLOUT AND RAINOUT ARE INCLUDED.

265 RSAC RSAC GENERATES A FISSION PRODUCT INVENTORY FROM A GIVEN SET OF REACTOR OPERATING CONDITIONS AND THEN COMPUTES THE EXTERNAL GAMMA DOSE, THE DEPOSITION GAMMA DOSE, AND/OR THE INHALATION-INGESTION DOSE TO CRITICAL BODY ORGANS AS A RESULT OF EXPOSURE TO THESE FISSION PRODUCTS. PROGRAM OUTPUT INCLUDES REACTOR OPERATING HISTORY, FISSION PRODUCT INVENTORY, DOSAGES, AND INGESTION PARAMETERS.

268 TRANS-FUGUE1 TRANS-FUGUE1 IS A TRANSIENT SINGLE CHANNEL, TWO-PHASE FLOW, AND HEAT TRANSFER CODE FOR ANALYSIS OF POSTULATED REACTOR INCIDENTS INVOLVING BOILING. THE CODE IS BASED ON A HOMOGENEOUS HYDRODYNAMIC MODEL WHICH ASSUMES EQUAL PHASE TEMPERATURES AND VELOCITIES. IT ASSUMES VAPOR GENERATION TO BE HEAT TRANSFER LIMITED ONLY, AND CALCULATES AXIAL DISTRIBUTION OF VELOCITY, VOID FRACTION, PRESSURE, COOLANT TEMPERATURE, AND FUEL TEMPERATURE. TRANSIENTS SIMULATING POWER EXCURSIONS, LOSS OF FLOW, LOSS OF PRESSURE AND CHANNEL PLUGGING CAN BE STUDIED.

278 WHAM/WATER-HAMMER WHAM IS USED TO CALCULATE PRESSURE, VELOCITY, AND FORCE TRANSIENTS IN THE LIQUID FILLED PIPING NETWORKS. IT CAN BE APPLIED TO MULTI-LOOP, COMPLEX PIPING NETWORKS CONSISTING OF ANY NUMBER OF DEAD ENDS, ELBOWS, ORIFICES, MULTIPLE BRANCH TEES, CHANGES OF FLOW PASSAGE CROSS SECTIONS, CHECK VALVES, PUMPS, PRESSURIZERS OR TANKS, AND EXIT VALVES. HYDRAULIC LOSSES ARE CONSIDERED. TRANSIENTS CAN BE INITIATED EITHER, BY CLOSURE OR BY OPENING OF ONE OR MORE EXIT VALVES (OR BY THE SYSTEM RUPTURE AT ONE OR MORE PLACES), OR BY A PRESCRIBED CHANGE IN THE TANK (OR PRESSURIZER) PRESSURE, OR BY THE PUMP POWER FAILURE. WATER-HAMMER ANALYZES HYDRAULIC TRANSIENTS IMPOSED BY THE SUDDEN DISCHARGE OF LIQUID FROM THE PRIMARY COOLANT LOOP, BEFORE ANY GENERATION OF VAPOR HAS TAKEN PLACE. EFFECTS OF HYDRAULIC LOSSES ARE NEGLECTED, THE LOOP PRESSURIZER IS ASSUMED TO REMAIN CONSTANT AT INITIAL PRESSURE AND THE PRESENCE OF A PUMP IS IGNORED. IT WAS DESIGNED FOR THE EARLY BLOWDOWN ANALYSIS FOR LOFT.

R284 M0555 (ACT1)

THIS PROGRAM WILL PREDICT THE PRESSURE TRANSIENT IN THE CONTAINMENT VESSEL AS A FUNCTION OF TIME AS A RESULT OF A MAJOR RUPTURE IN THE PRIMARY SYSTEM OF A PRESSURIZED WATER REACTOR. MEANS ARE AVAILABLE FOR INTRODUCING WATER AND HEAT INTO THE CONTAINMENT VESSEL. THERE ARE ALSO PROVISIONS FOR SPECIFYING VARIOUS TYPES OF HEAT SINKS FOR ENERGY ABSORPTION. THE TEMPERATURE DISTRIBUTION IN THESE HEAT SINKS IS CALCULATED AS A FUNCTION OF TIME. DISCHARGE RATES OF THE CONTAINMENT MAY ALSO BE DETERMINED AS A FUNCTION OF BOTH TIME AND THE INTERNAL PRESSURE.

R295 FLASH2

FLASH2 DETERMINES THE TRANSIENT RESPONSE OF A WATER REACTOR TO A LOSS-OF-COOLANT ACCIDENT OR SEVERE VARIABLE PRESSURE OPERATION.

297 CONTEMPT

THE CONTEMPT COMPUTER PROGRAM PREDICTS THE PRESSURE-TEMPERATURE RESPONSE OF A DRY WELL CONTAINMENT BUILDING TO A LOSS-OF-COOLANT ACCIDENT. THE CONTAINMENT VOLUME IS SEPARATED INTO A LIQUID REGION AND A VAPOR REGION. EACH REGION IS ASSUMED TO HAVE A UNIFORM TEMPERATURE BUT THE TEMPERATURE OF THE TWO REGIONS MAY BE DIFFERENT. THE CONTAINMENT BUILDING IS REPRESENTED AS CONSISTING OF SEVERAL HEAT CONDUCTING STRUCTURES WHOSE THERMAL BEHAVIOR CAN BE DESCRIBED BY THE ONE-DIMENSIONAL MULTIREGION HEAT CONDUCTION EQUATION. WATER AND ENERGY RATES FROM DISCHARGE OF COOLANT, THE BOILING OF RESIDUAL WATER BY REACTOR DECAY HEAT, THE SUPERHEATING OF STEAM PASSING THROUGH THE CORE, AND METAL-WATER REACTIONS ARE ASSUMED AVAILABLE FROM PREVIOUS CALCULATIONS AND ARE INPUT DATA TO CONTEMPT. PROGRAM OUTPUT INCLUDES CONTAINMENT VOLUME PRESSURE AND TEMPERATURES, TEMPERATURE THROUGH BUILDING STRUCTURES, AND THE AMOUNT OF WATER, VAPOR, AND ENERGY IN THE CONTAINMENT VOLUMES. THE PROGRAM ALSO CALCULATES BUILDING LEAKAGE AND THE EFFECTS OF ENGINEERED SAFEGUARD DEVICES SUCH AS WATER SPRAY AND FAN COOLER SYSTEMS.

328 NURLOC-1.0

NURLOC-1.0 PERFORMS CORE/PRESSURE VESSEL THERMAL ANALYSIS FOR A NUCLEAR REACTOR LOSS-OF-COOLANT ACCIDENT.

330 ECCSA1

ECCSA1 PREDICTS THE THERMAL AND HYDRAULIC PERFORMANCE OF REACTOR CORE CHANNELS DURING A LOSS-OF-COOLANT ACCIDENT AND SUBSEQUENT EMERGENCY CORE COOLING INJECTION.

H. STEADY-STATE AND TRANSIENT HEAT TRANSFER

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152 ARGUS (RE248) THIS PROGRAM CALCULATES TRANSIENT TEMPERATURES IN A CONCENTRIC, CYLINDRICAL CONFIGURATION. UP TO 25 CONCENTRIC REGIONS ARE ALLOWED, EACH CONTAINING EITHER A STATIONARY (SOLID OR NON-FLOWING LIQUID) OR TURBULENTLY FLOWING (LIQUID OR GAS) MATERIAL. ANY STATIONARY REGION CAN HAVE SPATIAL- AND TIME-DEPENDENT HEAT GENERATION. TEMPERATURES ARE CALCULATED AT NODE POINTS EQUALLY-SPACED WITHIN A REGION. FILM COEFFICIENTS ON FLOWING REGION BOUNDARIES ARE CALCULATED BY THE PROGRAM. TIME-DEPENDENT COOLANT VELOCITIES ARE PERMITTED. THE HEAT SOURCE IS ASSUMED TO BE ANGULAR INDEPENDENT. AXIAL HEAT CONDUCTION IS NEGLECTED, BUT AXIAL HEAT TRANSPORT DUE TO MATERIAL MOTION IS CONSIDERED IN THE FLOWING REGIONS.

155 PTH1 THIS PROGRAM CALCULATES CONTAINMENT SHELL PRESSURE-TEMPERATURE HISTORY RESULTING FROM AQUEOUS COOLANT SYSTEM BLOWDOWN.

182 AXFLU AXFLU CALCULATES HEAT TRANSFER FROM A LATTICE OF FLUID CYLINDRICAL FUEL PINS TO COOLANT IN EITHER SLUG OR FULLY-DEVELOPED LAMINAR FLOW. SOLUTION IS ANALYTIC AND CLOSED, EXCEPT FOR SATISFYING BOUNDARY CONDITIONS AT A FINITE, (BUT ARBITRARY), NUMBER OF BOUNDARY POINTS.

183 AXTHRM AXTHRM CALCULATES HEAT TRANSFER FROM A TRIANGULAR LATTICE OF CLAD SOLID CYLINDRICAL FUEL PIN TO COOLANT IN EITHER SLUG OR FULLY-DEVELOPED LAMINAR FLOW. SOLUTION IS ANALYTIC AND CLOSED, EXCEPT FOR SATISFYING BOUNDARY CONDITIONS AT A FINITE, (BUT ARBITRARY), NUMBER OF BOUNDARY POINTS.

198 HEATING2 HEATING2 IS A GENERALIZED HEAT TRANSFER CODE CAPABLE OF SOLVING TRANSIENT AND/OR STEADY-STATE COORDINATE SYSTEMS. THE SIMPLIFIED INPUT MAKES IT A VERY USEFUL CODE FOR THOSE PROBLEMS HAVING GEOMETRICAL CONFIGURATIONS WHICH CAN BE DESCRIBED BY PARALLEL AND PERPENDICULAR LINES OR PLANES, OR CONCENTRIC CIRCLES. THESE FIGURES CAN BE BROKEN UP INTO A MAXIMUM OF 100 ONE-MATERIAL REGIONS WITH POSITION AND TIME-DEPENDENT VOLUMETRIC HEAT GENERATION RATES. UP TO 40 MATERIALS WITH CONSTANT PROPERTIES CAN BE DESCRIBED, AND THE INITIAL TEMPERATURES OF THESE REGIONS ARE POSITION-DEPENDENT. THE BOUNDARIES OF THESE REGIONS CAN BE CONTACT, INSULATED, TIME-DEPENDENT TEMPERATURE CONTROLLED, OR FORCED CONVECTION WITH A TIME-DEPENDENT SINK TEMPERATURE. ALSO A RADIATION BOUNDARY WITH A TIME-DEPENDENT SINK TEMPERATURE IS INCLUDED.

205 BLCOST5 BLCOST5 COMBINES A REACTOR KINETICS CODE WITH A TWO-DIMENSIONAL HEAT TRANSFER CODE, AND IS APPLICABLE TO PROBLEMS FOR WHICH THE SPACE-INDEPENDENT FORM OF THE REACTOR KINETICS EQUATION IS APPLICABLE.

242 RATH MESHER/RATH WANTON THESE CODES SOLVE THE TWO- AND THREE-DIMENSIONAL HEAT CONDUCTION PROBLEM IN GENERAL GEOMETRIES. TIME, TEMPERATURE, AND POSITION-DEPENDENT PHYSICAL PROPERTIES CAN BE USED.

246 FLOW-MODEL A MULTI-CHANNEL, TWO-DIMENSIONAL, TWO-PHASE FLOW MODEL, DESIGNED TO COMPUTE THE AXIAL AND RADIAL COOLANT DENSITY AND QUALITY PROFILES, THE AXIAL PRESSURE PROFILE AND THE WEIGHT FLOW DISTRIBUTION FOR AN OPEN MATRIX FLOW, BOILING WATER REACTOR.

256 MANTA MANTA IS A PROGRAM WHICH PROVIDES A THERMAL-HYDRAULIC NODAL ANALYSIS IN THE STEADY STATE. IT WAS DESIGNED TO ANALYZE FUEL ELEMENT CONFIGURATION IN THE SUPER-HEAT DEVELOPMENT PROGRAM. MANTA ANALYZES MIXING BETWEEN COOLANT CHANNELS, ALLOWS FOR TEMPERATURE VARIANT CONDUCTIVITY IN ADMITTANCE CALCULATION, AND MULTIPLE STACKED SEGMENTS THROUGH THE FUEL REGION FOR A 7 ELEMENT CLUSTER ANALYSIS OVER A LENGTH OF UP TO 8 FEET. MANTA IS DESIGNED FOR SINGLE-PASS STEAM FLOW. THE FLOW DIRECTION IN THE COOLANT CHANNELS MAY BE EITHER UP OR DOWN, THEREBY PERMITTING THE ANALYSIS OF TWO-PASS AS WELL AS SINGLE-PASS FUEL ELEMENTS. MANTA ACCOUNTS FOR THE HEAT TRANSFER AND PRESSURE DROP THAT MAY OCCUR BETWEEN COOLANT CHANNELS DUE TO MIXING AS WELL AS TO THE CONVENTIONAL HEAT TRANSFER AND PRESSURE DROP RELATIONSHIPS DUE TO FRICTION, DISCONTINUITIES, ACCELERATION, CONVECTION, CONDUCTION, AND RADIATION. MANTA ALLOWS FOR THE CALCULATION AT EACH NODE OF THE MATERIAL PROPERTIES VISCOSITY, SPECIFIC HEAT, CONDUCTIVITY, AND SPECIFIC VOLUME TO CORRESPOND TO THE ACTUAL NODE TEMPERATURE BEING SOLVED FOR.

R267 WATER WATER IS A SUBROUTINE USED TO EXTRACT THERMODYNAMIC AND TRANSPORT PROPERTIES OF LIQUID, VAPOR, AND SUPERCRITICAL WATER BY TABULAR INTERPOLATION OVER THE RANGE OF STATES - 14.5 TO 14,500 PSIA AND 32 TO 1472 DEGREES F. THESE PROPERTIES ARE SPECIFIC VOLUME, SPECIFIC ENTHALPY, DYNAMIC VISCOSITY, AND THERMAL CONDUCTIVITY, TABULATED FOR PRESSURE AND TEMPERATURE CONDITIONS.

R272 FIGRO FIGRO CALCULATES THE ONE-DIMENSIONAL STEADY-STATE TEMPERATURE DISTRIBUTION AND TOTAL FUEL SWELLING FOR METAL-CLAD, AXISYMMETRIC, BULK-OXIDE CYLINDRICAL FUEL ELEMENTS. THE FUEL PELLET MAY BE SOLID, ANNULAR, OR CONTAIN TWO RADIAL ZONES. OXIDE FUEL THERMAL CONDUCTIVITY IS A FUNCTION OF TEMPERATURE, DEPLETION, AND POROSITY. FUEL SWELLING IS A FUNCTION OF TEMPERATURE, DEPLETION, INTERNAL HYDROSTATIC PRESSURE, AND FISSIONING RATE. EITHER THE CLAD SURFACE FLUX OR THE TEMPERATURE AT THE INSIDE RADIUS OF THE FUEL MAY BE SPECIFIED AS A BOUNDARY CONDITION FOR THE HEAT CONDUCTION EQUATION.

R286 HOT2

HOT2 IS A DIGITAL COMPUTER PROGRAM TO SOLVE TWO-DIMENSIONAL PLANE AND AXIALLY SYMMETRIC STEADY-STATE AND TRANSIENT HEAT CONDUCTION PROBLEMS WITH DIAGONAL BOUNDARIES AND INTERFACES. MESH SPACING (AT MOST 5000 POINTS) IS COMPLETELY VARIABLE. AS MANY AS 99 REGIONS ARE PERMITTED IN ORDER TO DESCRIBE SPATIAL VARIATIONS IN MATERIAL PROPERTIES, HEAT GENERATION RATES, AND BOUNDARY CONDITIONS. THE HEAT GENERATION RATE AND BOUNDARY CONDITIONS MAY VARY WITH TIME.

R294 M0899/HOH

BY MAKING CALLS ON A SUBROUTINE CALLED HOH, M0899 EDITS THERMODYNAMIC AND TRANSPORT PROPERTIES OF WATER OVER THE RANGE 14.5 TO 2538 PSIA AND UP TO 608 DEGREES FAHRENHEIT BELOW SATURATION AND 932 DEGREES FAHRENHEIT ABOVE SATURATION.

R299 LION

LION IS A DIGITAL COMPUTER PROGRAM WHICH WILL SOLVE THREE-DIMENSIONAL TRANSIENT AND STEADY-STATE TEMPERATURE DISTRIBUTION PROBLEMS. THE INPUT CONSISTS OF GEOMETRY, PHYSICAL PROPERTIES, BOUNDARY CONDITIONS, INTERNAL HEAT GENERATION RATES, AND COOLANT FLOW RATES AS A FUNCTION OF TIME. IN ADDITION TO SOLVING PROBLEMS OF HEAT CONDUCTION IN A STRUCTURE, LION CAN HANDLE FORCED CONVECTION, FREE CONVECTION, AND RADIATION OR A COMBINATION OF THESE AT THE SURFACE OF THE STRUCTURE. THE OUTPUT CONSISTS OF COMPLETE NODAL TEMPERATURE DISTRIBUTION ALONG WITH SURFACE FLUXES AND HEAT TRANSFER COEFFICIENTS. AN OPTION IS INCLUDED IN THE PROGRAM FOR DETERMINING THE MEAN TEMPERATURE IN ANY SPECIFIED SECTION OF THE STRUCTURE.

I. DEFORMATION AND STRESS DISTRIBUTION COMPUTATIONS, STRUCTURAL ANALYSIS AND ENGINEERING DESIGN STUDIES

6/68

48 FUGUE THE FUGUE PROGRAM COMPUTES STEADY-STATE WALL AND BULK FLUID TEMPERATURE, VOID FRACTION, AND LOCAL PRESSURE IN LIQUID-COOLED CLOSED CHANNELS IN WHICH THE HEATING RATE IS SPECIFIED. THE REQUIRED RELATIONSHIPS ARE EXPRESSED IN GENERAL, NON-DIMENSIONAL FORM AND COMBINED IN AN INTERNALLY CONSISTENT MANNER TO ALLOW PREDICTIONS FOR A VARIETY OF COOLANTS AND SPECIFIED OPERATING CONDITIONS.

R 80 SCR2 SOR2 SOLVES FOR THE FORCES, STRESSES, DEFLECTIONS, AND STRAINS IN THIN SHELLS OF REVOLUTION. THE SHELLS MAY BE GENERAL SURFACES OF REVOLUTION WITH VARIABLE THICKNESSES AND ELASTIC MODULII. THIS INCLUDES THE MORE FAMILIAR FORMS - THE CIRCULAR FLAT PLATE, CONE, CYLINDER, SPHERE, ELLIPSE, AND TORII WITH CIRCULAR OR ELLIPTICAL CROSS SECTIONS, FOR WHICH A SIMPLIFIED INPUT IS USED. THE AXISYMMETRIC LOADINGS CONSIDERED INCLUDE ARBITRARY DISTRIBUTIONS OF NORMAL, TANGENTIAL AND MOMENT SURFACE LOADINGS, AS WELL AS EDGE FORCES AND DEFLECTIONS. THE EFFECTS OF RADIAL AND AXIAL TEMPERATURE DISTRIBUTIONS, CENTRIFUGAL LOADING DUE TO ROTATION ABOUT THE AXIS AND VIBRATION ARE INCLUDED. THE ADDITIONAL EFFECTS OF MISALIGNMENT, LINE LOADS, AND ELASTIC SUPPORTS AT THE SHELL INTERSECTIONS ARE CONSIDERED.

109 4RESTRAINT PIPE STRESS CODE THIS PROGRAM EVALUATES A FOUR RESTRAINT PIPING SYSTEM DESIGNED FOR HIGH TEMPERATURE OPERATION.

112 CROCK CROCK SELECTS THE MINIMUM WEIGHT DESIGN FOR A SPACE POWER PLANT IN WHICH THE WASTE CYCLE HEAT IS RADIATED DIRECTLY TO SPACE FROM THE CONDENSER. IT ACCOUNTS FOR HEAT TRANSFER, FLUID FLOW, METEOROID PROTECTION, AND THE GEOMETRIC PROPERTIES OF A RADIATOR-CONDENSER.

114 SHOCK SHOCK SELECTS THE MINIMUM WEIGHT DESIGN FOR A HEAT-REJECTION SYSTEM FOR A SPACE POWER PLANT IN WHICH THE SENSIBLE HEAT LCST FROM A SINGLE-PHASE FLUID IS RADIATED TO SPACE. IT ACCOUNTS FOR HEAT TRANSFER, FLUID FLOW, METEOROID PROTECTION AND RADIATOR GEOMETRY.

190 WCPXPRT THE PURPOSE OF THE CODE IS TO CHOOSE THE LIGHTEST REACTOR RESULTING FROM ALL POSSIBLE COMBINATIONS OVER THE RANGE OF REACTOR INDEPENDENT VARIABLES SPECIFIED. THE ENGINEERING CALCULATIONS FOR EACH REACTOR DESIGN ARE BASED ON A LIQUID-COOLED CYLINDRICAL REACTOR HAVING PIN-TYPE FUEL ELEMENTS. REACTOR CONTROL IS ASSUMED TO BE EFFECTED BY MOVABLE SIDE REFLECTORS SURROUNDING THE CORE. NECESSARY PHYSICS DATA IS INPUT IN THE FORM OF SECOND-DEGREE EQUATIONS AS A FUNCTION OF THE INDEPENDENT VARIABLES.

230 PIZZA

THIS CODE USES A NUMERICAL METHOD TO CALCULATE THE DEFORMATION AND RESULTING STRESS DISTRIBUTION OF TWO-DIMENSIONAL BODIES UNDER PLANE STRESS CONDITIONS. PIZZA IS INTENDED AS A RAPID AND FLEXIBLE DESIGN TOOL IN WHICH THE MATERIAL CONSTANTS, DIMENSIONS, AND LOADING CONDITIONS MAY BE READILY ALTERED. THIS CODE IS PARTICULARLY APPLICABLE TO THE SOLUTION OF IRREGULARLY-SHAPED PLANE BODIES WHICH DO NOT HAVE EITHER AN EXACT OR AN APPROXIMATE MATHEMATICAL SOLUTION.

232 GAZELLE5

DETERMINATION OF GAS-COOLED FAST REACTOR CORE PROPORTIONS REQUIRED TO SATISFY SPECIFIED DESIGN CONSTRAINTS AND COMPUTATION OF RESULTING PERFORMANCE CHARACTERISTICS.

233 CORE

THE REACTOR CORE CONFIGURATION DESIGNED BY THE CORE PROGRAM CONSISTS OF CYLINDRICAL FUEL ELEMENTS ARRANGED ON AN EQUILATERAL PITCH SPACING WITH INTERNAL AND EXTERNAL COOLING. THE FUEL ELEMENT IS COMPOSED OF THREE CONCENTRIC GRAPHITE RINGS, AN INTERNAL COOLANT CHANNEL, AND A SPINE. THE COOLANT CHANNEL IS FORMED BY THE SPINE AND INNER GRAPHITE RING. THE FUEL IS CONTAINED IN THE MIDDLE GRAPHITE RING. CORE WILL DETERMINE THE NUMBER OF FUEL ELEMENTS OF A SPECIFIED DIAMETER AND LENGTH REQUIRED TO SATISFY A GIVEN CENTRAL FUEL TEMPERATURE. IT ALSO ADJUSTS THE PITCH SPACING REQUIRED TO PRODUCE A DESIRED PRESSURE DROP AS WELL AS PROVIDING A SPECIFIED INTERNAL COOLING RATE.

244 CYCLOPS1

CYCLOPS1 IS A PROGRAM FOR THE ANALYSIS OF THERMODYNAMIC SYSTEMS. IT IS A GENERAL PURPOSE PROGRAM THAT PERFORMS A HEAT AND MASS BALANCE FOR THERMODYNAMIC SYSTEMS COMPOSED OF PUMPS, TURBINES, SEPARATORS, HEAT EXCHANGERS, COMPRESSORS AND FLOW THROTTLING DEVICES. THESE COMPONENTS MAY BE CONNECTED IN ANY DESIRED WAY. THE THERMODYNAMIC FLUIDS THAT MAY BE USED ARE WATER, AIR, NITROGEN, AND PARA-HYDROGEN.

250 SAFE-PCRS

SAFE-PCRS NUMERICALLY DETERMINES THE STRESS AND STRAIN DISTRIBUTION WITHIN EITHER HOMOGENEOUS OR HETEROGENEOUS THICK-WALLED BODIES OF REVOLUTION. IT IS DESIGNED FOR THE ANALYSIS OF MULTI-MATERIAL AXISYMMETRIC COMPOSITE STRUCTURES SUCH AS REINFORCED AND/OR PRESTRESSED CONCRETE VESSELS. DEFORMATIONS MUST BE WITHIN THE ELASTIC LIMIT OF THE MATERIALS CONSIDERED AND ONLY BODIES OF REVOLUTION SUBJECTED TO AXISYMMETRIC LOADING CAN BE TREATED.

251 SAFE-AXISYM

SAFE-AXISYM IS A PROGRAM FOR THE ANALYSIS OF MULTI-MATERIAL AXISYMMETRIC COMPOSITE STRUCTURES. IT IS DESIGNED FOR THE ANALYSIS OF HETEROGENEOUS STRUCTURES SUCH AS REINFORCED AND/OR PRESTRESSED CONCRETE VESSELS. THE STRUCTURE IS ASSUMED TO BE LINEARLY ELASTIC, AND ONLY BODIES OF REVOLUTION SUBJECTED TO AXISYMMETRIC LOADING CAN BE TREATED.

252 SAFE-PLANE SAFE-PLANE IS APPLIED TO TWO-DIMENSIONAL STRUCTURES OF ARBITRARY GEOMETRY UNDER IN-PLANE LOADS. EITHER PLANE STRESS OR PLANE STRAIN CONDITIONS MAY BE IMPOSED. MECHANICAL AND/OR THERMAL LOADS ARE PERMITTED.

253 SAFE-SHELL SAFE-SHELL IS USED TO DESIGN AND ANALYZE AXISYMMETRIC THIN SHELL STRUCTURES OF ARBITRARY GENERATRICES UNDER AXISYMMETRIC MECHANICAL AND/OR THERMAL LOADING CONDITIONS. THE INTERSECTION OF TWO OR MORE SHELLS CAN BE TREATED.

R266 CYGRO2 CYGRO2 IS USED TO DETERMINE STRESSES AND STRAINS DURING STEADY-STATE AND TRANSIENT POWER OPERATION OF AN OXIDE-FUELED, METAL CLAD ROD-TYPE FUEL ELEMENT IN A PRESSURIZED ENVIRONMENT. MAJOR LOADING CONDITIONS INCLUDE FUEL SWELLING, FISSION GAS AND COCLANT PRESSURE, CLAD GROWTH AND DIFFERENTIAL THERMAL EXPANSION. THE APPLICATION FOR WHICH THE PROGRAM HAS BEEN DEVELOPED IS ZIRCALOY TUBES CONTAINING BULK OXIDE FUEL. AXIAL AND AZIMUTHAL SYMMETRY OF TEMPERATURE AND STRESSES IS ASSUMED.

R282 SEALSHL2 THE SEALSHL2 PROGRAM DETERMINES STRESSES, STRAINS, DEFLECTIONS, AND REACTIONS IN A THICK SHELL OF REVOLUTION WITH AXISYMMETRIC LOADING. THE LOADING CONSISTS OF A TEMPERATURE DISTRIBUTION, INSIDE AND OUTSIDE PRESSURE DISTRIBUTIONS, AND CIRCUMFERENTIAL FORCES AND MOMENTS APPLIED TO THE MIDDLE SURFACE. THE SHELL IS LINEAR-ELASTIC WITH TENSILE, BENDING, AND SHEAR STRAINS.

R283 M0552 M0552 SOLVES THE TRANSIENT RESPONSE PROBLEM OF LINEAR ELASTIC, LUMPED-MASS SYSTEMS SUBJECTED TO A UNIDIRECTIONAL FOUNDATION TRANSIENT THAT CAN BE EITHER A VELOCITY OR ACCELERATION TRANSIENT. NORMAL MODE THEORY IS USED AND THE INPUT TO THE PROGRAM CONSISTS OF THE MODE SHAPES, FREQUENCIES, AND FOUNDATION TRANSIENT. ELEMENT EFFECTS ARE ALSO EVALUATED AS A FUNCTION OF TIME. MODAL DAMPING COEFFICIENTS MAY BE SPECIFIED.

300 SAFE-CREEP SAFE-CREEP PERFORMS A VISCO-ELASTIC ANALYSIS OF PLANE OR AXISYMMETRIC COMPOSITE CONCRETE STRUCTURES WITH AGE AND TEMPERATURE DEPENDENT CREEP DATA.

315 WIREX WIREX PRODUCES WIRING LISTS CONTAINING ALL THE INFORMATION NORMALLY FOUND ON DETAILED WIRING DRAWINGS. THESE LISTS ARE PRESENTED TO THE ELECTRICIANS AS A JOB BOOK CONTAINING INFORMATION IN THE ORDER IN WHICH IT IS REQUIRED FOR LOGICAL PROCEDURE THROUGH A WIRING JOB. WIREX LISTS WERE USED EXTENSIVELY DURING INSTALLATION FOR THE UHTREX PROJECT.

R329 MC457 (PIPE)

PIPE PERFORMS AN ELASTIC STRESS ANALYSIS OF A 3-DIMENSIONAL PIPING STRUCTURE WITH THERMAL STRESSES, REDUNDANT LOOPS, AND CONCENTRATED LOADS.

332 SAFE-3D

SAFE-3D IS A FINITE ELEMENT PROGRAM FOR THE THREE-DIMENSIONAL ELASTIC ANALYSIS OF HETEROGENEOUS COMPOSITE STRUCTURES. THE PROGRAM USES THE FOLLOWING TYPES OF FINITE ELEMENTS - (1) TETRAHEDRAL ELEMENTS TO REPRESENT THE CONTINUUM, (2) TRIANGULAR PLANE STRESS MEMBRANE ELEMENTS TO REPRESENT INNER LINER OR OUTER CASE, AND (3) UNIAXIAL TENSION-COMPRESSION ELEMENTS TO REPRESENT INTERNAL REINFORCEMENT. THE STRUCTURE CAN BE OF ARBITRARY GEOMETRY AND HAVE ANY DISTRIBUTION OF MATERIAL PROPERTIES, TEMPERATURES, SURFACE LOADINGS, AND BOUNDARY CONDITIONS.

J. GAMMA HEATING AND SHIELD DESIGN PROBLEMS

6/68

45 GRACE1 GRACE1 IS A MULTIGROUP, MULTIREGION, GAMMA-RAY ATTENUATION PROGRAM DESIGNED PRIMARILY FOR COMPUTING GAMMA-RAY HEATING AND GAMMA-RAY DOSE RATES IN MULTIREGION FINITE OR SEMI-INFINITE SLAB SHIELDS. A DIFFERENT BUILDUP FACTOR MAY BE SPECIFIED FOR EACH SOURCE REGION CONSIDERED.

46 GRACE2 GRACE2 IS A MULTIGROUP, MULTIREGION, GAMMA-RAY ATTENUATION PROGRAM TO COMPUTE THE TOTAL DOSE RATE OR HEAT GENERATION RATE FROM EITHER A SPHERICAL OR A CYLINDRICAL SOURCE. THE SOURCE, WHICH MAY BE LOCATED IN EITHER THE CENTRAL REGION OF THE SYSTEM OR IN A CONCENTRIC SHELL REGION SURROUNDING IT, MAY BE UNIFORM, EXPONENTIAL, OR HAVE A POLYNOMIAL VARIATION IN THE RADIAL DIRECTION. IN THE CASE OF CYLINDRICAL GEOMETRY, IT MAY ALSO HAVE A POLYNOMIAL VARIATION IN THE AXIAL DIRECTION.

91 FARSE1A THE PROGRAM COMPUTES THE NEUTRON LEAKAGE FROM A SHIELD ANNULUS. THE REMOVAL CROSS SECTIONS INCORPORATE MULTISCATTERING EFFECTS. DOSE DEPOSIT AT THE TARGET MESH IS THEN DETERMINED FROM THE ANGULAR DISTRIBUTION OF THE LEAKAGE NEUTRONS, INTEGRATED OVER THE SHIELD SURFACE.

110 SCARF2 SCARF2 EVALUATES THE FIRST-ORDER APPROXIMATION OF THE FAST NEUTRON CURRENT AT THE PAY LOAD SURFACE DUE TO NEUTRONS WHICH SCATTER FROM THE RADIATOR FINS. IT USES SNAP GEOMETRY AND IS A COMPLEMENTARY PROGRAM TO FARSE.

111 SCAR1 SCAR1 PERFORMS SINGLE SCATTER RAY TRACING AND EVALUATES SCATTERING FROM A RING. IT IS COMPLEMENTARY TO FARSE.

123 LIPRECAN1 LIPRECAN1 IS A TWO-DIMENSIONAL MONTE CARLO PROGRAM TO COMPUTE THE PENETRATION AND ENERGY DEPOSITION OF NEUTRONS IN PURE HYDROGENOUS MEDIA. THE CODE OFFERS THREE POSSIBLE GEOMETRIES, I. E., CONICAL, CYLINDRICAL AND ONE-DIMENSIONAL INFINITE SLAB GEOMETRY (THIS LATTER OPTION TO THE EXTENT THAT AN INFINITE SLAB MAY BE APPROXIMATED BY A CYLINDER OF LARGE RADIUS). THE CODE MAY BE USED WITH A MONO-DIRECTIONAL POINT OR BEAM SOURCE AND IS MODIFIED EASILY TO HANDLE ANGULAR DISTRIBUTIONS. ISOTROPIC SCATTERING IN THE CENTER OF MASS SYSTEM IS ASSUMED. THE FOLLOWING RESULTS ARE TABULATED WITH EACH SUMMARY -

- (A) ENERGY DEPOSITION DISTRIBUTION,
- (B) PARTICLE DEPOSITION DISTRIBUTION,
- (C) PARTICLE LEAKAGE FRACTION,
- (D) PARTICLE ABSORPTION FRACTION,
- (E) THE FRACTION OF PARTICLES REACHING THE CUTOFF ENERGY WHICH HAVE NEITHER LEAKED NOR HAVE BEEN ABSORBED,
- (F) THE AVERAGE ENERGY PER PARTICLE LEAKING FROM THE SYSTEM,
- (G) THE AVERAGE ENERGY PER PARTICLE REMAINING IN THE SYSTEM AS HEAT GENERATION,
- (H) THE AVERAGE NUMBER OF COLLISIONS PER HISTORY,
- (I) THE TOTAL NUMBER OF HISTORIES CURRENTLY BEING SUMMARIZED,
- (J) THE AVERAGE ENERGY DEPOSIT PER HISTORY THROUGH ABSORPTION,
- (K) THE AVERAGE ENERGY OF PARTICLES ABSORBED.

141 RATRAP THE RATRAP PROGRAM COMPUTES THE DOSE RATE AT SPECIFIED SPATIAL POINTS ABOUT A SYSTEM OF SNAP GEOMETRY.

142 MORTIMER MORTIMER COMPUTES THE DOSE RATE AT SPECIFIED SPATIAL POINTS ABOUT A SYSTEM OF SNAP GEOMETRY.

143 MAC MAC PERFORMS SLAB GEOMETRY, MULTIGROUP NEUTRON AND GAMMA RAY PENETRATION ANALYSIS FOR A MULTI-REGION REACTOR SHIELD. THE CODE CALCULATES THE FOLLOWING INFORMATION AS A FUNCTION OF DISTANCE THROUGH A REACTOR SHIELD ASSEMBLY -
 NEUTRON FLUXES FOR UP TO 35 ENERGY GROUPS,
 NEUTRON DOSE RATES,
 THE APPROXIMATE NEUTRON SPECTRUM,
 GAMMA RAY FLUXES FOR 7 ENERGY GROUPS,
 TOTAL GAMMA DOSE RATE, WITH A BREAKDOWN OF THE CONTRIBUTION FROM SOURCES IN EACH REACTOR AND SHIELD REGION,
 THE APPROXIMATE GAMMA RAY SPECTRUM.

197 SHCE THE SHOE CODE MAKES USE OF THE METHOD OF STEEPEST DESCENT TO FIND THE DIMENSIONS OF A MINIMUM-WEIGHT, THREE-LAYER SHIELD IN SPHERICAL GEOMETRY. THE WEIGHT MINIMIZATION IS CARRIED OUT SUBJECT TO THE CONSTRAINING CONDITION OF A CONSTANT DOSE RATE AT SOME SELECTED POINT OUTSIDE THE SHIELD.

259 MUSCAT MUSCAT COMPUTES THE INCIDENT
SCATTERED NEUTRON CURRENTS AS A FUNCTION OF POSITION WITHIN (1)
THE CAVITY FORMED BY TWO TRUNCATED CONCENTRIC SPHERES, (2) THE
CAVITY BETWEEN TWO CONCENTRIC CIRCULAR CYLINDERS, OR (3) A CYLIN-
DRICAL CAVITY.

K. TOTAL SYSTEMS ANALYSIS

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273 THREDES

THREDES IS A SCIENTIFIC APPLICATIONS PROGRAMMING SYSTEM. INCORPORATED IN THIS SYSTEM ARE THE NECESSARY MODULES TO PERFORM PARAMETRIC DESIGN STUDIES OF THERMAL REACTORS INCLUDING THE THERMAL CELL HOMOGENIZATION (BAM - ACC ABSTRACT 108), THE FAST SPECTRUM CALCULATION (FORM - ACC ABSTRACT 51), REACTOR DIFFUSION THEORY (FOG - ACC ABSTRACT 28), AND ZERO-DIMENSIONAL BURNUP (KINDLE) CALCULATIONS. THESE MODULES CAN BE USED IN CONJUNCTION WITH ONE ANOTHER OR INDIVIDUALLY.

L. DATA PREPARATION

6/68

189 SNC
FOR DSN AND TDC.

CALCULATION OF SN CONSTANTS

193 CSP2A
THIS CODE PREPARES 100 GROUP FORWARD AND ADJOINT MACROSCOPIC MIXTURE CROSS SECTIONS FOR SN TYPE CODES. THE 100 GROUP MICROSCOPIC CROSS SECTIONS ARE TAKEN FROM THE CROSS SECTION TAPE WRITTEN BY THE CODE CSP1. THE OUTPUT OF CSP2A IS IN THE FORM OF TWO TAPES, ONE CONTAINING THE FORWARD AND THE OTHER THE ADJOINT MACROSCOPIC CROSS SECTIONS. THE CROSS SECTIONS ARE ALSO PRINTED, IF DESIRED.

194 CSP1
THIS CODE PREPARES 100 GROUP MICROSCOPIC CROSS SECTIONS FOR SN TYPE REACTOR CODES, USING THE GAM2 NUCLEAR DATA TAPE FOR THE CROSS SECTION DATA FOR THE FIRST 99 GROUPS. THE DATA FOR GROUP 100 IS PART OF THE PUNCHED CARD INPUT. THE 100 GROUP SN CROSS SECTIONS, PREPARED BY THE CODE, ARE WRITTEN ON TAPE AND PRINTED, IF DESIRED. THE CROSS SECTIONS ARE FOR THE ISOTROPIC OPTION OF THE SN CODES. THE DIAGONAL TRANSPORT APPROXIMATION HAS BEEN USED IN CALCULATING THE TRANSPORT AND IN-GROUP SCATTERING CROSS SECTIONS. THE CORRECTION TERM USED TO CALCULATE THESE TWO CROSS SECTIONS IS SIMPLY $1/3$ OF THE P_1 SCATTERING CROSS SECTION. THE NUMBER OF OUT-OF-GROUP SCATTERING TERMS IS OPTIONAL. WHEN LESS THAN THE NUMBER APPEARING ON THE GAM2 TAPE ARE REQUESTED, THE REMAINING TERMS, FOR A GIVEN SOURCE GROUP, ARE SUMMED AND ADDED TO THE LAST DOWN-SCATTERING TERM SAVED. WHEN CROSS SECTIONS ARE LISTED IN THE SN FORMAT, IT IS POSSIBLE THAT ONE OR MORE INTERMEDIATE DOWN-SCATTERING TERMS WILL BE ZERO. WHEN THIS SITUATION ARISES, THE CODE WILL REPLACE THESE INTERMEDIATE ZEROS WITH THE QUANTITY $1.0 \times 10^{-(20)}$.

218 GAVER
GAVER CALCULATES GROUP AVERAGED CROSS SECTIONS FROM POINTWISE CROSS SECTION DATA FOR THE GAM2 99 FINE-GROUP LIBRARY. THE INPUT DATA ARE USED TO OBTAIN A SET OF INTERPOLATED CROSS SECTIONS, WHERE 20 INTERPOLATED CROSS SECTIONS ARE OBTAINED FOR EACH FINE GROUP. WHERE INPUT DATA DO NOT COVER THE FULL RANGE OF A FINE GROUP, THE GROUP-AVERAGED CROSS SECTION IS SET TO ZERO.

234 DPC
DPC PREPARES INPUT DATA FOR TWO-DIMENSIONAL NEUTRONIC CALCULATIONS FROM ENGINEERING DATA. A GENERAL DESCRIPTION OF REGIONS, IN TERMS OF MULTIPLE SUBREGIONS, ALLOWS FOR EXPLICIT CONSIDERATION OF DETAILED STRUCTURAL FORMS. DURING THE TRANSFORMATION OF DATA, GENERAL THERMAL EXPANSIONS ARE INCLUDED. ALSO, CONSIDERABLE DATA CHECKING AND EVALUATION ARE PERFORMED. COMBINED WITH THE FLEXIBLE NARRATIVE INPUT, THESE FEATURES MAKE THE CODE USEFUL IN THE PREPARATION OF ERROR-FREE INPUT FOR COMPLEX PROBLEMS.

296 GRDWRK GRDWRK GENERATES AS PUNCHED
OUTPUT THE BASIC FINITE ELEMENT REFERENCE GRID WORK FOR THE SAFE
CODES. THIS GENERATED GRID CONSISTS OF TRIANGULAR ELEMENTS AND
NODES, UNIAXIAL ELEMENTS, SUCH AS REINFORCEMENT BARS, TENDONS, AND
ANCHORS, AND BIAXIAL MEMBRANES, SUCH AS ANY THIN SHELL OR LINER.
THE PUNCHED OUTPUT SERVE AS DIRECT INPUT DATA TO THE SAFE CODES.

M. DATA MANAGEMENT

6/68

159 MCMUS MOMUS IS THE PROGRAM USED TO CONSTRUCT AND MAINTAIN THE NEUTRON CROSS SECTION MASTER TAPE DEVELOPED UNDER THE ATOMICS INTERNATIONAL AUTOMATED CROSS SECTION PROGRAM. THE LIBRARY CONTAINS 21 ELEMENTS WITH DATA FOR 11 PARAMETERS IN THE RANGE 0.001 EV TO 10 MEV. MOMUS WILL PERFORM THE FOLLOWING TASKS -

- (A) MAKE THE BINARY MASTER TAPE FROM CARDS,
- (B) LIST SELECTED ELEMENTS,
- (C) UPDATE - ADD, CORRECT AND REPLACE ANY DATA,
- (D) MAKE A SHORT TAPE CONTAINING SELECTED ELEMENTS,
- (E) PROVIDE GRAPHICAL DISPLAY OF SELECTED DATA, AND
- (F) PUNCH MICROSCOPIC DATA.

181 XLIBIT USING THE ANL STANDARD CROSS SECTION DATA CARDS AS INPUT, THE PROGRAM CAN PERFORM THE FOLLOWING OPERATIONS -

- (A) PREPARE A LIBRARY TAPE CONSISTING OF THE CROSS SECTION DATA FOR SETS OF MATERIALS, INCLUDING A DIRECTORY GIVING THE SETS ON THE TAPE AND THE MATERIALS IN EACH SET,
- (B) GIVEN AN EXISTING LIBRARY TAPE, THE PROGRAM CAN PREPARE A DUPLICATE TAPE OR A MODIFIED TAPE, THE MODIFICATIONS CONSISTING OF THE ADDITION AND DELETION OF SETS, THE ADDITION AND DELETION OF MATERIALS WITHIN A SET, AND THE RE-ORDERING OF THE SETS ON THE TAPE,
- (C) ANY OF THE DATA ON A LIBRARY TAPE MAY BE PRINTED, AND THE PROGRAM WILL PUNCH THE CROSS SECTION DATA CARDS THAT WERE USED TO PLACE ANY OF THE MATERIALS OR SETS ON THE TAPE WITH ALTERED SET AND MATERIAL NAMES IF DESIRED.

206 UNPACK UNPACK RETRIEVES NEUTRON CROSS SECTION DATA AND RELATED INFORMATION FROM A SCISRS (BROOKHAVEN NATIONAL LABORATORY) DATA TAPE. THE FORMAT AND STRUCTURE OF THE SCISRS TAPE IS DESCRIBED IN BNL-883 BY FRIEDMAN AND PLATT, JULY 1964.

236 DFSR DATA FILE SERVICE ROUTINES (DFSR) ARE USED TO STORE, DISTRIBUTE, AND CHECK THE DATA ON THE EVALUATED NUCLEAR DATA FILE (ENDF). THERE ARE 6 ROUTINES -

- DFSR1 - STORES DATA ON MASTER LIBRARY TAPES.
- DFSR2 - PROCESSES REQUESTS FOR DATA BY FINDING THE APPROPRIATE DATA ON THE MASTER TAPES AND COPYING THEM ONTO THE REQUESTORS TAPE.
- DFSR3 - CORRECTS DATA ON THE MASTER TAPES.
- DFSR4 - COPIES/PRINTS/PUNCHES SELECTED DATA FROM THE MASTER TAPES.
- DFSR5 - PRINTS LIST OF DATA RECENTLY ADDED TO MASTER TAPES AND PREPARES A CROSS-REFERENCE LIST OF ALL DATA ON THE MASTER TAPES.
- DFSR6 - MAKES COPIES AND CHECKS THE MASTER TAPES.

271 CLIP

CLIP IS THE CROSS SECTION
LIBRARY PREPARATION AND MAINTENANCE PROGRAM FOR FORM AND THREDES.

133 WED THIS PROGRAM EDITS THE MAGNETIC TAPE PRODUCED BY W-DSN CALCULATING REACTION RATES BY ENERGY AND BY VOLUME WITH TOTALS. IT CAN ALSO PRODUCE REACTION RATES FOR FED-IN CROSS SECTIONS.

207 CROSSPLOT
FROM NEUTRON CROSS SECTION DATA.

210 DTX THE DTX CODE CALCULATES EFFECTIVE MACROSCOPIC, HOMOGENEOUS, GROUP CROSS SECTIONS WHICH ARE SPACE-AVERAGED OVER THE FLUXES AND CURRENTS PRE-CALCULATED IN A ONE-DIMENSIONAL NEUTRON TRANSPORT CODE SUCH AS DTK OR DSN. ISOTROPIC AND ANISOTROPIC CROSS SECTIONS MAY BE INCLUDED.

239 CPS CPS PROVIDES A GRAPHICAL
MEANS OF COMPARING EXPERIMENTAL CROSS SECTION VALUES OBTAINED
FROM THE SCISRS LIBRARY TAPE WITH AN OPTION TO INCLUDE READING IN
NEW EXPERIMENTAL OR CALCULATED VALUES.

C. EXPERIMENTAL DATA PROCESSING

6/68

154 CROC90 THE CROC90 CODE WAS DEVELOPED FOR USE IN THE DATA REDUCTION OF OUT-OF-PILE FLUID FLOW EXPERIMENTS ON THE ML-1 FUEL ELEMENTS. THE CODE IS SPECIFICALLY DESIGNED TO EVALUATE FRICTION FACTORS, ENTRANCE AND EXIT COEFFICIENTS, AND ORIFICE CALIBRATIONS FROM HYDRODYNAMIC DATA OBTAINED FROM SINGLE PHASE EXPERIMENTAL FLUID FLOW TESTS IN AXIAL FLOW DUCTS.

164 BURP1 THE PROGRAM CALCULATES ABSOLUTE TOTAL EFFICIENCY FOR MONOENERGETIC GAMMA RAY INTERACTIONS IN CYLINDRICAL SCINTILLATION DETECTORS. THE ABSOLUTE TOTAL EFFICIENCY IS DEFINED AS THE FRACTION OF SOURCE GAMMAS WHICH INTERACT AT LEAST ONCE WITH THE CRYSTAL DETECTOR. CALCULATIONS ARE MADE FOR THE POINT ISOTROPIC SOURCE LOCATED ALONG THE AXIS OF SYMMETRY FOR SOLID CYLINDRICAL CRYSTALS, WITH OR WITHOUT A COAXIAL CYLINDRICAL WELL.

165 BURP2 THE PROGRAM CALCULATES THE ABSOLUTE TOTAL EFFICIENCY FOR MONOENERGETIC GAMMA RAYS INTERACTING IN CYLINDRICAL SCINTILLATION DETECTORS. THE ABSOLUTE TOTAL EFFICIENCY IS DEFINED AS THE FRACTION OF SOURCE GAMMAS WHICH INTERACT AT LEAST ONCE WITH THE CRYSTAL DETECTOR. CALCULATIONS ARE MADE FOR HOMOGENEOUS, ISOTROPIC CIRCULAR DISK OR CYLINDRICAL VOLUME SOURCES. SOURCES MUST BE SYMMETRICAL WITH THE AXIS OF SYMMETRY FOR SOLID CYLINDRICAL CRYSTALS, WITH OR WITHOUT A COAXIAL CYLINDRICAL WELL. SOURCE ABSORPTION AND SCATTERING MAY BE INCLUDED FOR VOLUME SOURCES.

166 BURP3 THE PROGRAM CALCULATES ABSOLUTE TOTAL EFFICIENCY FOR MONOENERGETIC GAMMA RAY INTERACTIONS IN CYLINDRICAL SCINTILLATION DETECTORS. THE ABSOLUTE TOTAL EFFICIENCY IS DEFINED AS THE FRACTION OF SOURCE GAMMAS WHICH INTERACT AT LEAST ONCE WITH THE CRYSTAL DETECTOR. CALCULATIONS ARE MADE FOR POINT ISOTROPIC SOURCES LOCATED BOTH ON AND OFF THE AXIS OF SYMMETRY FOR SOLID CYLINDRICAL CRYSTALS, WITH OR WITHOUT A COAXIAL CYLINDRICAL WELL. THE EFFICIENCIES FOR POINTS LOCATED AT SPECIFIED RADIAL OFF-AXIS POSITIONS, FOR A SINGLE AXIAL SOURCE-CRYSTAL DISTANCE, ARE NORMALIZED TO THE ON-AXIS EFFICIENCY FOR THE SAME AXIAL DISTANCE. THE ABSOLUTE EFFICIENCY FOR THE ON-AXIS POINT IS GIVEN.

169 BURP4 BURP4 CALCULATES THE PHOTO-FRACTION FOR MONOENERGETIC GAMMA RAYS INTERACTING IN SOLID CYLINDRICAL SCINTILLATION DETECTORS. PHOTO-FRACTION IS DEFINED AS THE FRACTION OF INTERACTING SOURCE GAMMAS THAT ARE TOTALLY ABSORBED (INCLUDING SECONDARIES) IN THE CRYSTAL. ISOTROPIC SOURCE GEOMETRIES ALLOWED ARE ISOTROPIC POINTS (ON OR OFF-AXIS), DISKS, CYLINDRICAL VOLUMES. ALLOWED MONODIRECTIONAL SOURCES, NORMAL TO CRYSTAL FACE, ARE NARROW BEAM COLLIMATED TO CRYSTAL AXIS, BROAD BEAM ILLUMINATING ENTIRE CRYSTAL FACE, AND COLLIMATED BEAM OF ANY SPECIFIED DIAMETER.

170 BURP5 BURP5 CALCULATES THE PHOTO-FRACTION FOR MONOENERGETIC GAMMA RAYS INTERACTING IN WELL-TYPE CYLINDRICAL SCINTILLATION DETECTORS. PHOTO-FRACTION IS DEFINED AS THE FRACTION OF INTERACTING SOURCE GAMMAS THAT ARE TOTALLY ABSORBED (INCLUDING SECONDARIES) IN THE CRYSTAL. ISOTROPIC SOURCE GEOMETRIES ALLOWED ARE ISOTROPIC POINTS (ON OR OFF-AXIS), DISKS, CYLINDRICAL VOLUMES. ALSO ALLOWS NARROW MONODIRECTIONAL BEAM COLLIMATED TO CRYSTAL AXIS, INCIDENT NORMAL TO WELL BOTTOM.

248 CCINC COINCIDENCE COUNTING DATA ARE TREATED TO OBTAIN SPECIFIC DISINTEGRATION RATES, CHANNEL EFFICIENCIES AND COUNT RATES, WEIGHTED MEANS, AND ALL ASSOCIATED STANDARD ERRORS. CORRECTIONS ARE MADE FOR UNEQUAL DEADTIME LOSS IN EACH CHANNEL, COINCIDENCE RESOLVING TIME LOSSES, DECAY DURING COUNTING, DECAY FROM A REFERENCE TIME, AND BACKGROUND IN EACH OF THE THREE CHANNELS. INPUT VARIABLES INCLUDE SAMPLE IDENTIFICATION, START TIME FOR COUNTING, SAMPLE REFERENCE TIME, COUNTING INTERVAL, TOTAL NUMBER OF COUNTS IN TWO SINGLE CHANNELS AND ONE COINCIDENCE CHANNEL, AND NORMALIZING SAMPLE VOLUME. INPUT PARAMETERS CONSIST OF DEADTIMES OF EACH SINGLE CHANNEL, COINCIDENCE RESOLVING TIME FOR ALL THREE CHANNELS, DECAY CONSTANT, BACKGROUND COUNT RATE FOR ALL THREE CHANNELS, AND STANDARD ERRORS FOR EACH OF THE ABOVE PARAMETERS. OPTIONAL INPUT ALLOWS DATE, GROUP CLASSIFICATION, AND A 3-DIGIT USER CODE. OUTPUT CONTAINS CORRECTED SINGLE CHANNEL, COINCIDENCE, AND DISINTEGRATION RATES REFERRED TO START OF COUNT, SPECIFIC DISINTEGRATION RATE (COUNTS/SECOND/UNIT VOLUME OR WEIGHT) REFERRED TO REFERENCE TIME, WEIGHTED MEAN AND ERROR OF ANY NUMBER OF PROBLEMS IN A GROUP, EFFICIENCIES OF THE TWO INDEPENDENT DETECTOR CHANNELS, DATE, SUMMARY OF BACKGROUND VALUES USED, IDENTIFICATION NUMBER, COUNT START TIME, AND UPPER AND LOWER LIMIT (ONE STANDARD DEVIATION) OF EACH OF THE COMPUTED QUANTITIES.

258 EXPN EXPN ANALYZES PULSED NEUTRON DATA USING THE GARELIS-RUSSEL TECHNIQUE (REFERENCE 1). BASICALLY THE CODE COMPUTES THE PROMPT DECAY CONSTANT, ALPHA, AND THE PARAMETER ($K \cdot \text{BETA}/L$) FROM EXPERIMENTAL DATA, WHICH IS DIRECTLY EXTRACTED FROM A TIME ANALYZER STORAGE MEMORY AND READ ONTO A PUNCHED PAPER TAPE. THE ALPHA-DETERMINATION PART OF THE CODE WAS ORIGINALLY OBTAINED UNDER THE NAME EXPLICIT FROM KNOLLS ATOMIC POWER LABORATORY BUT HAS SINCE BEEN MODIFIED. THE CODE PROVIDES OPTIONS FOR A PRE-BURST OR A POST-BURST BACKGROUND ANALYSIS. THAT IS, THE PARAMETERS ALPHA AND ($K \cdot \text{BETA}/L$) ARE OBTAINED USING A BACKGROUND MEASURED PRIOR TO THE BURST OR MEASURED AFTER THE BURST.

311 BURNUP BURNUP CORRELATES HEAVY ELEMENT ISOTOPIC ANALYSIS WITH FISSION PRODUCT NEODYMIUM, URANIUM, AND PLUTONIUM CONCENTRATIONS IN AN IRRADIATED URANIUM FUEL FOR CALCULATION OF BURNUP (ATOM PER CENT FISSION AND MWD/MT). REACTOR PARAMETERS, INCLUDING EFFECTIVE NEUTRON ABSORPTION CROSS SECTIONS FOR ALL URANIUM AND PLUTONIUM ISOTOPES, CAPTURE-TO-FISSION RATIOS FOR U235, PU239, AND PU241, A TWO-GROUP DESCRIPTION OF THE NEUTRON SPECTRUM, THE AVERAGE NEUTRON TEMPERATURE, THE REACTOR FAST FISSION FACTOR, AND THE DISTRIBUTION OF THE SOURCES OF FISSION AMONG THE FISSIONABLE NUCLIDES ARE COMPUTED FROM THE EXPERIMENTAL DATA OBTAINED FROM THE MASS SPECTROMETRIC ANALYSIS OF URANIUM, PLUTONIUM, AND NEODYMIUM.

333 TOAD
ANALYZE GAMMA RAY SPECTRA.

TOAD IS USED TO PROCESS AND

P. GENERAL MATHEMATICAL AND COMPUTING SYSTEM ROUTINES

6/68

R 43 CURFIT CURFIT IS A COMPOSITE PROGRAM FOR FITTING EXPERIMENTAL DATA POINTS WITH DIFFERENT TYPES OF COMMON ANALYTIC CURVES. THERE ARE AT PRESENT FIVE FITS AVAILABLE -

- (1) POLYNOMIAL $Y = \text{SUMMATION OVER } I \text{ OF } A(I) \cdot (X^{**}(I))$
- (2) EXPONENTIAL $Y = A \cdot \text{EXP}(BX)$
- (3) COSINE $Y = A \cdot \text{COS}(B(X+C))$
- (4) SERIES OF CUBICS $Y = A(J) + B(J) \cdot X + C(J) \cdot (X^{**}2) + D(J) \cdot (X^{**}3)$
- (5) FOURIER SERIES

62 LCS ALAMOS LEAST SQUARES THIS PROGRAM PERFORMS LEAST SQUARES FITTING OF LINEAR OR NONLINEAR FUNCTIONS IN SEVERAL INDEPENDENT VARIABLES. THE PROGRAM WILL DETERMINE AN ESTIMATE OF A IN THE FUNCTION $Y=F(X,A)$ BY MINIMIZING THE SUM OF SQUARES. IN THE FUNCTION, X IS A VECTOR OF OBSERVED VARIABLES AND A IS A VECTOR OF PARAMETERS TO BE DETERMINED. IN THIS CONTEXT, A LINEAR FUNCTION IS ONE WHOSE PARTIAL DERIVATIVES WITH RESPECT TO THE ELEMENTS OF A ARE ALL INDEPENDENT OF A. A NONLINEAR FUNCTION HAS AT LEAST ONE OF THE ELEMENTS OF A APPEARING IN AT LEAST ONE OF THESE PARTIAL DERIVATIVES.

186 LAG1/LAG2 LAG IS A SINGLE PASS LOAD AND GO ASSEMBLER DESIGNED TO ACCEPT IBM7090 FLOCO II INSTRUCTIONS.

321 EXPALS THIS PROGRAM FITS BY LEAST SQUARES A FUNCTION WHICH IS A LINEAR COMBINATION OF REAL EXPONENTIAL DECAY FUNCTIONS. THE FUNCTION IS

$Y(K) = \text{SUMMATION OVER } J \text{ OF } A(J) \cdot \text{EXP}(-\text{LAMBDA}(J) \cdot K)$
VALUES OF THE INDEPENDENT VARIABLE (K) AND THE DEPENDENT VARIABLE Y(K) ARE SPECIFIED AS INPUT DATA. WEIGHTS MAY BE SPECIFIED AS INPUT INFORMATION OR SET BY THE PROGRAM ($W(K) = 1/Y(K)$).

324 FRANTIC FRANTIC IS DESIGNED TO PROCESS RAW COUNTING DATA AND TO FIT IN THE LEAST SQUARES SENSE THESE DATA TO THE MULTIPLE EXPONENTIAL GROWTH AND DECAY EQUATIONS. THE PROGRAM CAN BE USED FOR SUMS OF EXPONENTIALS WITH POSITIVE, NEGATIVE, OR ZERO EXPONENTS AND POSITIVE OR NEGATIVE COEFFICIENTS.

R327 DAFT1 DAFT1 IS A PROGRAM FOR WEIGHTED LEAST SQUARES FITTING OF 0.0253 EV NEUTRON DATA FOR FIS-SILE NUCLIDES. THE PROGRAM ALSO CARRIES OUT COMPUTATIONS RELEVANT TO DISCERNING OVERALL GOODNESS OF FIT, PARTICULARLY DEVIANT DATA, AND DATA WHOSE IMPROVEMENT WOULD LEAD TO LARGE REDUCTIONS IN ERROR OF EACH FITTED PARAMETER.

VII. KWIC INDEX

STANT CALC 0 TO 10 MEV	BNWL	1107	F4	RS	PLT=	GAMTEC2, MULTI-GP CON	185
FUSION SLAB CYL SPHERE	BNWL	1107	F4	RS	P T=	HFN, 1-D MULTI-GP DIF	241
ON FOR SAFE PROGRAMS	GGA	1108	F4	RS	P =	GRDWRK, GRID GENERATI	296
DIFFUSION XY RZ GEOMETRY	GA	1108	F+BAL	RS	P =	GAMBLES, 2-D MULTI-GP	222
ESS ANALYSIS, 2-D BODIES	GA	1108	F4	RS	P =	SAFE-PLANE, PLANE STR	252
ALYSIS THIN SHELLS	GA	1108	F4	RS	P =	SAFE-SHELL, STRESS AN	253
HIELDING CODE CAVITY GEOM	GA	1108	F4	RS	P T=	MUSCAT, VIEW FACTOR S	259
CIDENT ANALYSIS DOSE CALC	GA	1108	F4	RS	T=	GADOSE/DOSET, HTGR AC	261
ERING LAW CALCULATION	GA	1108	F4	RS	P =	GASKET, THERMAL SCATT	263
X-SECTION CALC MODERATOR	GGA	1108	F4	RS	P =	GAKER, INELASTIC SCAT	289
SIS REACTOR KINETICS EQNS	GGA	1108	F4	RS	P =	GASA, STABILITY ANALY	290
X-SECTIONS HEX LATTICE	GGA	1108	F4	RS	=	HEXSCAT, ELASTIC SCAT	291
ESONANCE PARAMETER CALC	GGA	1108	F4	RS	P =	PSEUDO, STATISTICAL R	292
IONS FAST THERMAL SPECTRA	GGA	1108	F4	RS	PLT=	GGC4, MULTI-GP X-SECT	298
TIC ANALYSIS CONCRETE	GGA	1108	F4	RS	P =	SAFE-CREEP, VISCOELAS	300
C FISSION PRODUCT RELEASE	GGA	1108	F4	RS	P =	FREVAP6, HTGR METALLI	301
EL CYCLE CALCULATION	GGA	1108	F4	S	T=	GAFFE, EQUILIBRIUM FU	302
ETICS 2-D HEAT TRANSFER	GGA	1108	F4	RS	P T=	BLOOST6, COMBINED KIN	303
ON XY RZ RTHETA GEOMETRY	BNWL	1108	F4	RS	P =	PERT4, 2-D PERTURBATI	304
E FAST REACTOR X-SEC CALC	BNWL	1108	F4	RS	P T=	FCC4, FUNDAMENTAL MOD	306
TIME-DEPENDENT DIFFUSION	GGA	1108	F4	RS	P T=	GAKIN, 1-D MULTIGROUP	310
GROUP-AVERAGED X-SEC CALC	GGA	1108	F4	RS	P T=	GAF/GAR, SPECTRA AND	316
ENDENT REACTOR KINETICS	GGA	1108	F4	RS	P =	GAPOTKIN, SPACE-INDEP	317
P DIFFUSION AND DEPLETION	GGA	1108	F4	RS	P T=	FEVER7, 1-D MULTIGROU	318
ER DISTRIBUTION SEARCH	GGA	1108	F4	RS	P T=	GASP7, 1-D BURNUP POW	319
OEFFICIENT CALCULATION	GGA	1108	F4	RS	P T=	TEMCO7, TEMPERATURE C	320
IFFUSION AND DEPLETION	BNWL	1108	F4	RS	P T=	2DB, 2-D MULTIGROUP D	325
E STRUCTURE STRESS STUDY	GGA	1108	F4	RS	P T=	SAFE-3D, 3-D COMPOSIT	332
NALYZER GAMMA-RAY SPECTRA	GGA	1108	F4	RS	P T=	TOAD, PROCESSING OF A	333
RS = WATER, STEAM TABLES	14.5-14,500	PSIA	32-472	DEGF	BAPL 2000 F4		R267
P T= M0899/HOH, STEAM TABLES	14.5-2538	PSIA			BAPL 6600 F4	RS	R294
ACH INELASTIC SCATTERING	CDC	1604	F63	RS	P =	HAFEVER, HAUSER-FESHB	14
SION SLAB CYLINDER SPHERE	CDC	1604	F63	RS	P T=	FOG, 1-D FEW-GP DIFFU	28
FFUSION SLAB CYL SPHERE	CDC	1604	F63	RS	PLT=	AIM6, 1-D MULTI-GP DI	29
N FOR AIM AND FOG CODES	CDC	1604	F63	RS	=	PERT, 1-D PERTURBATIO	30
P DIFFUSION XYZ GEOMETRY	CDC	1604	F63	RS	P T=	WHIRLAWAY, 3-D 2-GROU	32
ECTRUM X-SECTION CALC	CDC	1604	F63	RS	PLT=	GAM1, FAST NEUTRON SP	33
P DIFFUSION SLAB CYLINDER	CDC	1604	F63	RS	P T=	20GRAND, 2-D FEW-GROU	40
ENUATION SLAB GEOMETRY	CDC	1604	F63	RS	P =	GRACE1, GAMMA-RAY ATT	45
ENUATION CYL SPHERE GEOM	CDC	1604	F63	RS	P =	GRACE2, GAMMA-RAY ATT	46
RATE FROM A CLOUD	CDC	1604	F63	RS	P =	CLOUD, GAMMA-RAY DOSE	47
TRON SPECTRUM X-SECTIONS	CDC	1604	F63	RS	PLT=	TEMPEST2, THERMAL NEU	50
ECTRUM X-SECTION CALC	CDC	1604	F63	RS	PLT=	FORM, FAST NEUTRON SP	51
1-GP S4 APPROXIMATION	CDC	1604	F63	RS	=	S4 CYL CELL CODE, 1-D	53
P DIFFUSION DEPLETION	CDC	1604	F63	RS	LT=	SIZZLE, 1-D MULTIGROU	58
DIFFUSION CYLINDER SLAB	CDC	1604	F63	RS	P T=	EQUIPOISE3A, 2-D 2-GP	87
GRAL X-SECTION CALC	CDC	1604	F63	RS	PLT=	ARES2, RESONANCE INTE	89
FFUSION SLAB CYL SPHERE	CDC	1604	F63	RS	PLT=	FAIM, 1-D MULTI-GP DI	120
LIDE CHAIN EQUATIONS	ORNL	1604	F63	RSBP	=	NUCY, SOLUTION OF NUC	134
ISCRETE ORDINATE CALC	UNC	1604	F63	RS	PLT=	DTF, 1-D MULTIGROUP D	144
I-GP SN APPROXIMATION RZ	PW	1604	F63	RS	P T=	FORTTRAN TDC, 2-D MULT	161
CREATE ORDINATE CODE	UNC-LASL	1604	F63	RS	=	2DF, 2-D MULTI-GP DIS	173
LD FROM REACTION OR DECAY	ORNL	1604	F63	RS	P =	ISOTOPES, MAXIMUM YIE	179
ECAY CHAIN ANALYSIS	ORNL	1604	F63	RS	P =	ISOCRUNCH, REACTION D	180
OCO2 INSTRUCTION SET	PW	1604	F+CDP	RS	P T=	LAG, ASSEMBLER FOR FL	186
IMPELLER DESIGN STUDY	PW	1604	F63	RS	P T=	PIP, CENTRIFUGAL PUMP	187
CTION EVALUATION	PW	1604	F63	RS	P =	CMPXMAT, TRANSFER FUN	188
N CONSTANTS FOR DSN TDC	PW	1604	F63	RS	P =	SNC, CALCULATION OF S	189
GHT OPTIMIZATION STUDY	PW	1604	F63	RS	=	WOPEXPRT, REACTOR WEI	190
ONICS-HYDRODYNAMICS SPH	PW	1604	F63	RS	P =	AX-TNT, COUPLED NEUTR	191
LUX SYNTHESIS PROGRAM CYL	PW	1604	F63	RS	T=	2DXYL, 3-D MULTI-GP F	192

IBRARY TAPE PREPARATION	PW	1604 F63	RS	=	CSP2A, SN X-SECTION L	193
BRARY TAPE PREPARATION	PW	1604 F63	RS	=	CSP1, SN X-SECTION LI	194
ON, GAMMA-RAY HISTORIES	PW	1604 F+CDP	RS P T=	FMC-N/FMC-G, MC NEUTR		195
TDC OR 2DXY FLUX INPUT	PW	1604 F63	RS	=	TDP, 2-D PERTURBATION	199
NNEL X-SEC EVALUATION	ORNL	1604 F63	RS P T=	JUPITOR1, COUPLED-CHA		308
ODUCTION FLUX, X-SEC CALC	ORNL	1604 F63	RS P	=	ISOSEARCH, ISOTOPE PR	322
RING PHASE-SHIFT ANALYSIS	ORNL	1604 F63	RS P	=	PEGGY, ELASTIC SCATTE	334
FUNCTION RADIAL INTEGRALS	ORNL	1604 F63	RS P	=	RAMES, PARTICLE WAVE	335
ION CALC FROM DSN OUTPUT	PW	1604 LAG1	RS P	=	DTX, EFFECTIVE X-SECT	210
N APPROX SLAB CYL SPHERE	PW	1604 LAG1	RS P	=	MGDSN, 1-D MULTI-GP S	211
AI 7090 F2	RS	=	SAIL, 1-C	1-GP SN APPROXIMATION SLAB GEOMETRY		52
RS	=	S4 CYL CELL CODE, 1-D	1-GP S4 APPROXIMATION	AI 7090 F2		53
3 RS	=	S4 CYL CELL CODE, 1-D	1-GP S4 APPROXIMATION	CDC 1604 F6		53
APL 6600 F4	RS P T=	WIGL2, 1-D	2-GP SPACE-TIME DIFFUSION 3-GEOM	B	R274	
OA 360 F4	RS PLT=	FIRE2, 1-D	AGE-DIFFUSION SLAB CYLINDER SPHERE	R		9
UK-R 7090 F+FAP	RSBPLT=	CRAM, 1-D	AND 2-D MULTI-GP DIFFUSION PROGRAM			103
GA 1108 F4	RS P T=	GASP7, 1-D	BURNUP POWER DISTRIBUTION SEARCH	G		319
AI 7090 F2	RS P T=	FOG, 1-D	FEW-GP DIFFUSION SLAB CYLINDER SPHERE			28
CDC 1604 F63	RS P T=	FOG, 1-D	FEW-GP DIFFUSION SLAB CYLINDER SPHERE			28
PP 7090 F2	RS P	=	MIST, 1-D FEW-GP SN DOUBLE SN APPROX SLAB GEOM			59
A 7090 F2	RSB T=	FEVER, 1-D	FEW-GP DIFFUSION DEPLETION PROGRAM	G		117
P 7040 F4	RS P T=	TOPIC, 1-D	FEW-GP SN APPROXIMATION CYLINDER	P		148
P 7090 F2	RS P T=	TOPIC, 1-D	FEW-GP SN APPROXIMATION CYLINDER	P		148
A 7044 F4	RS P T=	GASP2, 1-D	FEW-GP DIFFUSION POWER DIST SEARCH	G		220
4 F4	RS P T=	RELOAD-FEVER, 1-D	FEW-GP DIFFUSION DEPLETION	GA 704		221
A 7044 F4	RS P T=	TEMCO, 1-D	FEW-GP DIFFUSION TEMP COEF CALC	G		225
360 F4	RSBP T=	THREDES, 1-D	FEW-GP DIFFUSION DESIGN SYSTEM	AI		273
AI 7090 F+FAP	RS PLT=	AIM6, 1-D	MULTI-GP DIFFUSION SLAB CYL SPHERE			29
CDC 1604 F63	RS PLT=	AIM6, 1-D	MULTI-GP DIFFUSION SLAB CYL SPHERE			29
7090 F2	RS LT=	SIZZLE, 1-D	MULTIGROUP DIFFUSION DEPLETION	AI		58
C 1604 F63	RS LT=	SIZZLE, 1-D	MULTIGROUP DIFFUSION DEPLETION	CD		58
360 F4	RSBP T=	SIZZLE, 1-D	MULTIGROUP DIFFUSION DEPLETION	AI		58
90 FLOCO	RSBP T=	GE-HAPO-S13, 1-D	MULTI-GP DOUBLE SN APPROX	BNWL 70		75
I 7090 F+FAP	RS LT=	ULCER, 1-D	MULTI-GP DIFFUSION SLAB CYL SPHERE	A		118
AI 7090 F+FAP	RS LT=	FAIM, 1-D	MULTI-GP DIFFUSION SLAB CYL SPHERE			120
CDC 1604 F63	RS PLT=	FAIM, 1-D	MULTI-GP DIFFUSION SLAB CYL SPHERE			120
EC 360 F4	RS P T=	FAIMOS, 1-D	MULTI-GP DIFFUSION SLAB CYL SPHERE	BO		120
K-W 7090 F2	RS P T=	W-DSN, 1-D	MULTI-GP SN APPROX SLAB CYL SPHERE	U		132
UNC 1604 F63	RS PLT=	DTF, 1-D	MULTIGROUP DISCRETE ORDINATE CALC			144
AI 7090 F2	RS PLT=	DTF2, 1-D	MULTI-GP DISCRETE ORDINATE CALC			151
I 360 F4	RSBPLT=	ANISN, 1-D	MULTI-GP DISCRETE ORDINATE CALC	A		151
LASL 7030 F4	RS P T=	DTF4, 1-D	MULTI-GP DISCRETE ORDINATE PROGRAM			209
LRC 7094 F4	RS P T=	DTF4, 1-D	MULTI-GP DISCRETE ORDINATE PROGRAM			209
LASL 6600 F4	RS P T=	DTF4, 1-D	MULTI-GP DISCRETE ORDINATE PROGRAM			209
BEC 625 F4	RS P T=	DTF4, 1-D	MULTI-GP DISCRETE ORDINATE PROGRAM			209
ANL 360 F4	RS P T=	DTF4, 1-D	MULTI-GP DISCRETE ORDINATE PROGRAM			209
W 1604 LAG1	RS P	=	MGDSN, 1-D MULTI-GP SN APPROX SLAB CYL SPHERE	P		211
BNWL 1107 F4	RS P T=	HFN, 1-D	MULTI-GP DIFFUSION SLAB CYL SPHERE			241
NL 3600 F63	RS P T=	MACH1, 1-D	MULTI-GP DIFFUSION SLAB CYL SPHERE	A		262
7030 F4	RS P T=	DTF-BURN, 1-D	MULTI-GP DTF4 WITH DEPLETION	LASL		269
F4	RSBPLT=	CAESAR4/LIBLST, 1-D	MULTI-GP DIFFUSION + LIB	AI 360		270
3600 F36	RS P T=	SNARG-1D, 1-D	MULTI-GP DISCRETE ORDINATE CALC	ANL		288
GA 1108 F4	RS P T=	GAKIN, 1-D	MULTIGROUP TIME-DEPENDENT DIFFUSION	G		310
A 1108 F4	RS P T=	FEVER7, 1-D	MULTIGROUP DIFFUSION AND DEPLETION	GG		318
AI 7090 F2	RS	=	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES			30
CDC 1604 F63	RS	=	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES			30
BOEC 360 F4	RS P	=	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES			30
7090 F2	RS	=	SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY	AI		52
RS	=	S4 CYL CELL CODE, 1-D	1-GP S4 APPROXIMATION	AI 7090 F2		53
RS	=	S4 CYL CELL CODE, 1-D	1-GP S4 APPROXIMATION	CDC 1604 F63		53
90 F2	RS	=	SNAPKIN5/5A, 1-REGION KINETICS SNAP GEOMETRY	AI 70		122

2, MULTI-GP CCNSTAT CALC 0 TO 10 MEV	BNWL 7090 F+FAP RS PLT=	GAMTEC	185
2, MULTI-GP CCNSTAT CALC 0 TO 10 MEV	BNWL 1107 F4 RS PLT=	GAMTEC	185
ON BNWL 1108 F4 RS P T=	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETI		325
UNC-LASL 1604 F63 RS =	2DF, 2-D MULTI-GP DISCRETE ORDINATE CODE		173
TDP, 2-D PERTURBATION TDC OR	2DXY FLUX INPUT PW 1604 F63 RS =		199
M CYL PW 1604 F63 RS T=	2DXYL, 3-D MULTI-GP FLUX SYNTHESIS PROGRA		192
OM AGN 7090 FLOCO RSBP =	2DXY, 2-D MULTI-GP SN APPROXIMATION XY GE		18
LASL 7094 FAP RS P T= RATH,	2- OR 3-D HEAT CONDUCTION LUMPED MASS		242
090 F2 RS P T= EQUIPOISE3,	2-D 2-GROUP DIFFUSION SLAB CYL ORNL 7		39
90 F2 RS P T= EQUIPOISE3A,	2-D 2-GP DIFFUSION CYLINDER SLAB ORNL 70		87
04 F63 RS P T= EQUIPOISE3A,	2-D 2-GP DIFFUSION CYLINDER SLAB CDC 16		87
= FLOW-MODEL, MULTI-CHANNEL	2-D 2-PHASE FLOW AI 7094 F2 RS P		246
IZZA, PLANE STRESS ANALYSIS OF	2-D BODIES GA 7044 F4 RS P T= P		230
-PLANE, PLANE STRESS ANALYSIS,	2-D BODIES GA 1108 F4 RS P = SAFE		252
T= DPC, DATA PREPARATION FOR	2-D DESIGN PROGRAMS LASL 7090 F2 RS P		234
APL 6600 F4 RS T= MO807,	2-D DIFFUSION ABSORPTION REMOVAL X-SECS B		R280
APDA 7094 F4 RS P = MARS,	2-D EXCURSION CALCULATION R-Z GEOMETRY		293
LRL 709 F2 RS = FIRN,	2-D FEW-GP S4 APPROXIMATION RZ GEOMETRY		7
L 7090 F2 RS P T= 20GRAND,	2-D FEW-GROUP DIFFUSION SLAB CYLINDER ORN		40
1604 F63 RS P T= 20GRAND,	2-D FEW-GROUP DIFFUSION SLAB CYLINDER CDC		40
GA 7090 F+FAP RSBP T= ODB,	2-D FEW-GP DIFFUSION BURNUP RZ GEOMETRY		99
IBM 360 F+BAL RS P T= PDQ5,	2-D FEW-GROUP DIFFUSION AND DEPLETION		R336
BLCOST5, POINT-KINETICS WITH	2-C HEAT TRANSFER GA 7044 F4 RS T=		205
T= 8LOOST6, COMBINED KINETICS	2-D HEAT TRANSFER GGA 1108 F4 RS P		303
360 F4 RS P T= HWOCR-SAFE,	2-D MONTE CARLO CELL CALCULATION AI		307
AGN 7090 FLOCO RSBP = 2DXY,	2-D MULTI-GP SN APPROXIMATION XY GEOM		18
0 F+FAP RSBPLT= CRAM, 1-D AND	2-D MULTI-GP DIFFUSION PROGRAM UK-R 709		103
0 F+FAP RS P T= EXTERMINATOR,	2-D MULTI-GP DIFFUSION PROGRAM ORNL 709		156
F4 RS P T= EXTERMINATOR2,	2-D MULTI-GP DIFFUSION PROGRAM ORNL 360		156
04 F63 RS P T= FORTRAN TDC,	2-D MULTI-GP SN APPROXIMATION RZ PW 16		161
-LASL 1604 F63 RS = 2DF,	2-D MULTI-GP DISCRETE ORDINATE CODE UNC		173
P RS P T= VARI-QUIR, TIME-DEP	2-D MULTI-GP DIFFUSION WAL 7090 F+FA		212
7044 F+MAP RSBP T= GAMBLE4,	2-C MULTI-GP DIFFUSION XY RZ GEOMETRY GA		222
1108 F+BAL RS P = GAMBLE5,	2-D MULTI-GP DIFFUSION XY RZ GEOMETRY GA		222
L 7090 F+FAP RS P T= ASSAULT,	2-D MULTI-GP DIFFUSION DEPLETION CODE ORN		240
094 F4 RS P T= VARI-QUIR3,	2-D MULTI-GP DIFFUSION XY RZ RTH WAL 7		264
ESJ 635 F4 RS PLT= BISYN,	2-C MULTI-GP DIFFUSION SYNTHESIS CALC G		287
LRC 7090 F+MAP RS P T= TDSN,	2-D MULTIGROUP DISCRETE ORDINATE PROGRAM		312
BNWL 1108 F4 RS P T= 2DB,	2-D MULTIGROUP DIFFUSION AND DEPLETION		325
PW 1604 F63 RS = TDP,	2-D PERTURBATION TDC OR 2DXY FLUX INPUT		199
NWL 1108 F4 RS P = PERT4,	2-D PERTURBATION XY RZ RTHETA GEOMETRY B		304
BAPL 6600 F4 RS P T= HOT2,	2-D TRANSIENT HEAT CONDUCTION PROGRAM		R286
2 RS P T= EQUIPOISE3A, 2-D	2-GP DIFFUSION CYLINDER SLAB ORNL 7090 F		87
63 RS P T= EQUIPOISE3A, 2-D	2-GP DIFFUSION CYLINDER SLAB CDC 1604 F		87
6600 F4 RS P T= WIGL2, 1-D	2-GP SPACE-TIME DIFFUSION 3-GEOM BAPL		R274
F2 RS P T= WHIRLAWAY, 3-D	2-GROUP DIFFUSION XYZ GEOMETRY ORNL 7090		32
F63 RS P T= WHIRLAWAY, 3-D	2-GROUP DIFFUSION XYZ GEOMETRY CDC 1604		32
F2 RS P T= EQUIPOISE3, 2-D	2-GROUP DIFFUSION SLAB CYL ORNL 7090		39
FLOW-MODEL, MULTI-CHANNEL 2-D	2-PHASE FLOW AI 7094 F2 RS P =		246
-RAFF, RESONANCE INTEGRAL CALC	2-REG CELL ANL 3600 F63 RSBP T= RIFF		213
IONS AI 7094 F2 RS P T=	2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECT		254
IONS ANL 3600 F63 RSBP T=	2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECT		254
INDER ORNL 7090 F2 RS P T=	20GRAND, 2-D FEW-GROUP DIFFUSION SLAB CYL		40
INDER CDC 1604 F63 RS P T=	20GRAND, 2-D FEW-GROUP DIFFUSION SLAB CYL		40
CURSION CALCULATIONS GESJ	2000 F4 RS P T= FORE, FAST REACTOR EX		174
TEGRAL X-SECTION CALC GESJ	2000 F2 RS P = RAPTURE, RESONANCE IN		176
ANCE X-SECTION CALC GESJ	2000 F2 RS P = DOPIE, RESOLVED RESON		177
RAGED DOPPLER EFFECTS GESJ	2000 F2 RS P = SPARTA, SPATIALLY-AVE		178
4.5-14,500PSIA 32-472DEGF BAPL	2000 F4 RS = WATER, STEAM TABLES 1		R267
ANCE INTEGRAL CALCULATION BW	2000 F4 RS = STRIP, RESOLVED RESON		305
, STEAM TABLES 14.5-14,500PSIA	32-472DEGF BAPL 2000 F4 RS = WATER		R267

ON SLAB CYLINDER SPHERE	ROA	360 F4	RS PLT=	FIRE2, 1-D AGE-DIFFUSI	9
FOR AIM AND FOG CODES	BOEC	360 F4	RS P =	PERT, 1-D PERTURBATION	30
RATE FROM A CLOUD	SRL	360 F4	RS P =	CLOUD, GAMMA-RAY DOSE	47
RCN SPECTRUM X-SECTIONS	BOEC	360 F4	RS PLT=	TEMPEST2, THERMAL NEUT	50
DIFFUSION DEPLETION	AI	360 F4	RSBP T=	SIZZLE, 1-D MULTIGROUP	58
DIFFUSION SLAB CYL SPHERE	BOEC	360 F4	RS P T=	FAIMOS, 1-D MULTI-GP D	120
ENT KINETICS W/FEEDBACK	ROA	360 F4	RS P =	AIREK3, SPACE-INDEPEND	121
ELASTIC SCAT RESONANCES	AI	360 F4	RS PLT=	AILMOE, X-SECTION CALC	147
SCRETE ORDINATE CALC	AI	360 F4	RSBPLT=	ANISN, 1-D MULTI-GP DI	151
TI-GP DIFFUSION PROGRAM	ORNL	360 F4	RS P T=	EXTERMINATOR2, 2-D MUL	156
TE REQUIREMENT STUDY	AI	360 F4	RS P T=	AISITE2, PARAMETRIC SI	172
CURSION CALCULATIONS	ANL	360 F4	RS P T=	FORE2, FAST REACTOR EX	174
CRETE ORDINATE PROGRAM	ANL	360 F4	RS P T=	DTF4, 1-D MULTI-GP DIS	209
LTI-GP DIFFUSION + LIB	AI	360 F4	RSBPLT=	CAESAR4/LIBLST, 1-D MU	270
LIBRARY UTILITY ROUTINE	AI	360 F4	RSB LT=	CLIP, FORM OR THREDES	271
FFUSION DESIGN SYSTEM	AI	360 F4	RSBP T=	THREDES, 1-D FEW-GP DI	273
IMENT ANALYSIS SYSTEM	BNL-SRL	360 F4	RS PLT=	HAMMER, CRITICAL EXPER	277
LOWDOWN ANALYSIS LOFT	UG	360 F4	RS P =	WATER-HAMMER, LIQUID B	278
CARLO CELL CALCULATION	AI	360 F4	RS P T=	HWOCR-SAFE, 2-D MONTE	307
NT REACTOR KINETICS	AI	360 F4	RS P T=	TSN, SPATIALLY-DEPEND	309
DUCTION FLUX, X-SEC CALC	ORNL	360 F4	RS P =	ISOSEARCH, ISOTOPE PRO	322
REACTOR DYNAMICS	AI	360 F+8AL	RS P T=	AIROS2A, SIMULATION OF	326
FFUSION AND DEPLETION	IBM	360 F+8AL	RS P T=	PDQ5, 2-D FEW-GROUP DI	R336
ECTRUM X-SECTION CALC	ANL	3600 F63	RS LT=	GAM1, FAST NEUTRON SP	33
CS-HYDRODYNAMICS SPHERE	CDC	3600 F63	RSBP =	AX1, COUPLED NEUTRONI	102
NSFER FUNCTION SYNTHESIS	ANL	3600 F63	RSBP T=	TRAFICORPORATION, TRA	135
LTIGROUP SLAB GEOMETRY	ANL	3600 F63	RSBPLT=	MAC, SHIELD DESIGN MU	143
ERATURE CALC CYLINDER	ANL	3600 F63	RSBP T=	ARGUS, TRANSIENT TEMP	152
AND POWER DISTRIBUTION	CDC	3600 F63	RSBP T=	FLARE, 3-D REACTIVITY	167
NT INVERSE KINETICS CALC	ANL	3600 F63	RSBP =	R102, SPACE-INDEPEND	168
LEUS X-SECTION CALC	ANL	3600 F63	RSBP =	NEARREX, COMPOUND NUC	171
REQUIREMENT STUDY	ANL	3600 F63	RS P T=	R153, PARAMETRIC SITE	172
RARY UTILITY ROUTINE	ANL	3600 F63	RSBP T=	XLIBIT, X-SECTION LIB	181
SPECTRUM X-SECTION CALC	ANL	3600 F63	RS P T=	THERMOS(ANL), THERMAL	184
INTEGRAL CALC 2-REG CELL	ANL	3600 F63	RSBP T=	RIFF-RAFF, RESONANCE	213
INTEGRAL CALC HOMOGENEOUS	ANL	3600 F63	RSBP =	MISH-MASH, RESONANCE	214
UNTING DATA REDUCTION	ANL	3600 F63	RSBP =	COINC, COINCIDENCE CO	248
OPTICAL MODEL X-SECTIONS	ANL	3600 F63	RSBP T=	2PLUS, NON-SPHERICAL	254
NT KINETICS KEX OPTIONS	ANL	3600 F63	RSBP =	R101, SPACE-INDEPEND	255
DIFFUSION SLAB CYL SPHERE	ANL	3600 F63	RS P T=	MACH1, 1-D MULTI-GP D	262
IPING SYSTEM ANALYSIS	KE	3600 F4	RSBP =	WHAM, LIQUID-FILLED P	278
ONANCE ABSORPTION, CELL	ANL	3600 F36	RSBP T=	RABBLE/WLIB/FLAT, RES	281
P DISCRETE ORDINATE CALC	ANL	3600 F36	RS P T=	SNARG-1D, 1-D MULTI-G	288
7090 F2	RS P T=	WHIRLAWAY, 3-D	2-GROUP DIFFUSION XYZ GEOMETRY	ORNL	32
1604 F63	RS P T=	WHIRLAWAY, 3-D	2-GROUP DIFFUSION XYZ GEOMETRY	CDC	32
1108 F4	RS P T=	SAFE-3D, 3-D	COMPOSITE STRUCTURE STRESS STUDY	GGA	332
00 F+ASC RS	T=	PDQ7, 1,2 OR 3-D	FEW-GP DIFFUSION DEPLETION	BAPL 66	R275
094 FAP	RS P T=	RATH, 2- OR 3-D	HEAT CONDUCTION LUMPED MASS	LASL 7	242
W 1604 F63	RS T=	2DXYL, 3-D	MULTI-GP FLUX SYNTHESIS PROGRAM CYL P		192
ESJ 635 F4	RS P T=	FLARE, 3-D	REACTIVITY AND POWER DISTRIBUTION	G	167
DC 3600 F63	RSBP T=	FLARE, 3-D	REACTIVITY AND POWER DISTRIBUTION	C	167
KAPL 6600 F4	RS P T=	LION, 3-D	TEMPERATURE DISTRIBUTION PROGRAM		R299
1-D 2-GP SPACE-TIME DIFFUSION	3-GEOM	BAPL 6600 F4	RS P T=	WIGL2,	R274
LC AI 7090 F+FAP RS	T=	4RESTRAINT PIPE STRESS, MAXIMUM MOMENT CA			109
CRETE ORDINATE PROGRAM	BEC	625 F4	RS P T=	DTF4, 1-D MULTI-GP DIS	209
SECTION CALCULATION	GEC	625 F4	RS P =	AVOID, ANNULAR VOID X-	276
AND POWER DISTRIBUTION	GESJ	635 F4	RS P T=	FLARE, 3-D REACTIVITY	167
CURSION CALCULATIONS	GESJ	635 F4	RS P T=	FORE2, FAST REACTOR EX	174
ERMAL-HYDRAULIC ANALYSIS	GESJ	635 F4	RS P T=	MANTA, STEADY-STATE TH	256
CE EPITHERMAL X-SECTIONS	GESJ	635 F4	RS P T=	REAX, RESOLVED RESONAN	257
ED NEUTRON SOURCE DATA	GESJ	635 F4	RS P T=	EXPAN, ANALYSIS OF PULS	258

FFUSION SYNTHESIS CALC	GESJ	635 F4	RS PLT=	BISYN, 2-D MULTI-GP DI	287
ISOTOPIC BURNUP ANALYSIS	GEV	635 F4	RS P T=	BURNUP, HEAVY ELEMENT	311
COOLANT THERMAL ANALYSIS	BMI	6400 F4	RS P T=	NURLOC-1.0, LOSS-OF-C	328
NT + EMERGENCY COOLING	BMI	6400 F4	RS P T=	ECCSA1, LOSS-OF-COOLA	330
EXPERIMENTAL DATA POINTS	KAPL	6600 F+ASC	RS P T=	CURFIT, CURVE FITTING	R 43
SHELLS OF REVOLUTION	KAPL	6600 F4	RS P T=	SOR2, STRESS ANALYSIS	R 80
SCRETE ORDINATE PROGRAM	LASL	6600 F4	RS P T=	DTF4, 1-D MULTI-GP DI	209
IS CYL FUEL ELEMENT	BAPL	6600 F4	RS T=	CYGR02, STRESS ANALYS	R266
LING TEMPERATURE STUDY	BAPL	6600 F4	RS P T=	FIGRO, LSBR FUEL SWEL	R272
-TIME DIFFUSION 3-GEOM	BAPL	6600 F4	RS P T=	WIGL2, 1-D 2-GP SPACE	R274
GP DIFFUSION DEPLETION	BAPL	6600 F+ASC	RS T=	PDQ7, 1,2 OR 3-D FEW-	R275
ABSORPTION REMOVAL X-SECS	BAPL	6600 F4	RS T=	M0807, 2-D DIFFUSION	R280
ESS ANALYSIS AXISYM LOAD	BAPL	6600 F4	RS P T=	SEALSHL2, SHELL STR	R282
IS LINEAR ELASTIC SYSTEMS	BAPL	6600 F4	RS P T=	M0552, DYNAMIC ANALYS	R283
COOLANT ACCIDENT ANALYSIS	BAPL	6600 F4	RS P T=	M0555(ACT1), LOSS-OF-	R284
ONANCE INTEGRAL HEX CELL	BAPL	6600 F4	RS P T=	RESQ2/RESQ0/DBF1, RES	R285
EAT CONDUCTION PROGRAM	BAPL	6600 F4	RS P T=	HOT2, 2-D TRANSIENT H	R286
ES 14.5-2538 PSIA	BAPL	6600 F4	RS P T=	M0899/HOH, STEAM TABL	R294
NT ACCIDENT ANALYSIS	BAPL	6600 F4	RS P T=	FLASH2, LOSS-OF-COOLA	R295
DISTRIBUTION PROGRAM	KAPL	6600 F4	RS P T=	LION, 3-D TEMPERATURE	R299
DEPLETION FISSION PRODUCT	BAPL	6600 F4	RS P =	CINDER(M0102), POINT	313
FIT FISSION NUCLIDE DATA	BAPL	6600 F4	RS P T=	DAFT1, LEAST SQUARES	R327
STRESS OF PIPING SYSTEM	BAPL	6600 F4	RS P T=	M0457(PIPE), ELASTIC	R329
W TRANSIENT ANALYSIS	BAPL	6600 F4	RS P T=	FLOT1(M0219), PWR FLO	R331
SCRETE ORDINATE PROGRAM	LASL	7030 F4	RS P T=	DTF4, 1-D MULTI-GP DI	209
P DTF4 WITH DEPLETION	LASL	7030 F4	RS P T=	DTF-BURN, 1-D MULTI-G	269
CS WITH 2-D HEAT TRANSFER	GA	7044 F4	RS T=	BLOOST5, POINT-KINETI	205
M SCISRS X-SECTION TAPE	GA	7044 F+MAP	RSBP =	UNPACK, RETRIEVAL FRO	206
TS FROM X-SECTION TAPES	GA	7044 F4	RSBP T=	CROSSPLOT, SC4020 PLO	207
OM B-W RESONANCE DATA	GA	7044 F4	RS P =	FASDOP, X-SECTIONS FR	216
EF CALC FOR ANGULAR DIST	GA	7044 F4	RS P =	LEGCOEF3, LEGENDRE CO	217
L X-SECTION AVERAGING	GA	7044 F4	RS P =	GAVER, ENERGY INTERVA	218
LAP AND LATTICE EFFECTS	GA	7044 F+MAP	RS P T=	GAROL, RESONANCE OVER	219
FUSION POWER DIST SEARCH	GA	7044 F4	RS P T=	GASP2, 1-D FEW-GP DIF	220
-GP DIFFUSION DEPLETION	GA	7044 F4	RS P T=	RELOAD-FEVER, 1-D FEW	221
DIFFUSION XY RZ GEOMETRY	GA	7044 F+MAP	RSBP T=	GAMBLE4, 2-D MULTI-GP	222
LE ANALYSIS W/REFUELING	GA	7044 F4	RS P T=	REVISED-GAD, FUEL CYC	223
STS PERFORMANCE STUDY	GA	7044 F4	RS P =	WAMPUM, FUEL CYCLE CO	224
FUSION TEMP COEF CALC	GA	7044 F4	RS P T=	TEMCO, 1-D FEW-GP DIF	225
FORMANCE AND PRICE STUDY	GA	7044 F+MAP	RS P T=	OPUS, POWER PLANT PER	226
OR DESIGN CRITERIA COSTS	GA	7044 F4	RS P T=	STMGEN, STEAM GENERAT	227
ENT PRESSURE POST RUPTURE	GA	7044 F4	RS T=	PRECON, HTGR CONTAINM	228
TRESS ANALYSIS STUDY	GA	7044 F4	RS P T=	ELBOW, FUEL ELEMENT S	229
NALYSIS OF 2-D BODIES	GA	7044 F4	RS P T=	PIZZA, PLANE STRESS A	230
ODUCT ACTIVITY DIST STUDY	GA	7044 F4	RS P =	RAD2, HTGR FISSION PR	231
CORE THERMAL DESIGN STUDY	GA	7044 F4	RS P =	GAZELLE5, GAS-COOLED	232
ION FUEL TEMPERATURE CODE	GA	7044 F4	RS =	CORE, CORE CONFIGURAT	233
AW X-SECTION CALCULATION	GA	7044 F4	RS =	FLANGE1, SCATTERING L	247
YSIS AXISYMMETRIC LOAD	GA	7044 F4	RS P T=	SAFE-PCRS, STRESS ANA	250
NALYSIS AXISYMMETRIC LOAD	GA	7044 F4	RS P T=	SAFE-AXISYM, STRESS A	251
ANALYSIS PARTIAL REFUEL	GA	7044 F4	RS P T=	GARGOYLE, FUEL CYCLE	260
APPROXIMATION CYLINDER	PP	7040 F4	RS P T=	TOPIC, 1-D FEW-GP SN	148
MULTI-LEVEL CONVOLUTION	PP	7040 F4	RS P =	COMBCO, COMBINED B-W	203
FETY ANALYSIS PROGRAM	PP	7040 F+MAP	RS P T=	RSAC, RADIOLOGICAL SA	265
LANT ACCIDENT ANALYSIS	PPC	7040 F+MAP	RS P T=	CONTEMPT, LOSS-OF-COO	297
PROXIMATION RZ GEOMETRY	LRL	709 F2	RS =	FIRN, 2-D FEW-GP S4 AP	7
ENERAL CURVE FITTING	LASL	7094 F4	RS =	LASL LEAST SQUARES, G	62
ED NEUTRON-HYDRODYNAMICS	APDA	7094 F2	RS P =	WEAK EXPLOSION, COUPL	145
LOWING-DOWN DENSITY CALC	AI	7094 F+FAP	RS PLT=	TYCHE3, MONTE CARLO S	149
RE TEMPERATURE HISTORY	KE	7094 F2	RSBP T=	PTH1, BLOWDOWN PRESSU	155
ARY UTILITY PROGRAM	AI	7094 F+FAP	RS LT=	MOMUS, X-SECTION LIBR	159

G X-SECTIONS PARAMETERS	AI	7094	F+FAP	RS	PLT=	GRAVE, GROUP-AVERAGIN	162
ENT KINETICS W/FEEDBACK	AI	7094	F+MAP	RS	P T=	AIROS, SPACE-INDEPEND	163
NT INVERSE KINETICS CALC	WAL	7094	F4	RS	=	R102, SPACE-INDEPEND	168
ITE REQUIREMENT STUDY	AI	7094	F+FAP	RS	P T=	AISITE2, PARAMETRIC S	172
MOLTEN FUEL TUBE BUNDLES	LASL	7094	F2	RS	P =	AXFLU, HEAT TRANSFER	182
SOLID FUEL TUBE BUNDLES	LASL	7094	F2	RS	P =	AXTHRM, HEAT TRANSFER	183
D, EXTERNAL+INTERNAL DOSE	AI	7094	F+FAP	RS	PLT=	CURIE/DOSE/THUNDERHEA	196
PTIMIZATION DOSE CALC	AI	7094	F2	RS	P =	SHOE, SHIELD WEIGHT O	197
STEADY-STATE HEAT TRANSFER	AI	7094	F2	RS	P T=	HEATING2, TRANSIENT S	198
BLOWDOWN ANALYSIS LOFT	KE	7094	F+MAP	RSBP	T=	SATURATED BLOWDOWN1,	200
AND X-SECTION CALCULATION	GEV	7094	F+FAP	RS	P T=	EPITHERMOS, SPECTRUM	201
GRAL X-SECTION CALC	AI	7094	F+MAP	RS	PLT=	TRIX1, RESONANCE INTE	208
SCRETE ORDINATE PROGRAM	LRC	7094	F4	RS	P T=	DTF4, 1-D MULTI-GP DI	209
ALC FOR ANGULAR DIST DATA	AI	7094	F+MAP	RS	P =	CHAD, LEGENDRE COEF C	215
-SECTIONS FOR GAMMA-RAYS	GEC	7094	F4	RS	P =	GAMMA-P, PRODUCTION X	235
B-W RESONANCE PARAMETERS	WAL	7094	F2	RS	P =	EXT, X-SECTIONS FROM	238
M SCISRS X-SECTION TAPES	WAL	7094	F2	RS	T=	CPS, SC4020 PLOTS FRO	239
CONDUCTION LUMPED MASS	LASL	7094	FAP	RS	P T=	RATH, 2- OR 3-D HEAT	242
IC CYCLE ANALYSIS	AI	7094	F2	RS	PLT=	CYCLOPS1, THERMODYNAM	244
NNEL 2-D 2-PHASE FLOW	AI	7094	F2	RS	P =	FLOW-MODEL, MULTI-CHA	246
WITH BURNUP CYL LATTICE	WAPD	7094	F+MAP	RS	PLT=	LASER, SPECTRUM CALC	249
OPTICAL MODEL X-SECTIONS	AI	7094	F2	RS	P T=	2PLUS, NON-SPHERICAL	254
NT KINETICS KEX OPTIONS	WAL	7094	F4	RS	P =	R101, SPACE-INDEPEND	255
-GP DIFFUSION XY RZ RTH	WAL	7094	F4	RS	P T=	VARI-QUIR3, 2-D MULTI	264
NT FLOW AND HEAT TRANSFER	AI	7094	F2	RS	P T=	TRANS-FUGUE1, TRANSIE	268
ALCULATION R-Z GEOMETRY	APDA	7094	F4	RS	P =	MARS, 2-D EXCURSION C	293
ENT REACTOR KINETICS	AI	7094	F4	RS	P T=	TSN, SPATIALLY-DEPEND	309
GAMMA-RAY RADIOACTIVITY	IITR	7094	F4	RSBP	LT=	NAP, NEUTRON-INDUCED	314
EXPONENTIAL DECAY CURVES	LRL	7094	F2	RS	P =	EXPALS, LEAST SQUARES	321
APPROXIMATION XY GEOM	AGN	7090	FLOCO	RSBP	=	2DXY, 2-D MULTI-GP SN	18
SION SLAB CYLINDER SPHERE	AI	7090	F2	RS	P T=	FOG, 1-D FEW-GP DIFFU	28
FFUSION SLAB CYL SPHERE	AI	7090	F+FAP	RS	PLT=	AIM6, 1-D MULTI-GP DI	29
N FOR AIM AND FOG CODES	AI	7090	F2	RS	=	PERT, 1-D PERTURBATIO	30
P DIFFUSION XYZ GEOMETRY	ORNL	7090	F2	RS	P T=	WHIRLAWAY, 3-D 2-GROU	32
ECTRUM X-SECTION CALC	GA	7090	F2	RS	PLT=	GAM1, FAST NEUTRON SP	33
UP DIFFUSION SLAB CYL	ORNL	7090	F2	RS	P T=	EQUIPOISE3, 2-D 2-GRO	39
P DIFFUSION SLAB CYLINDER	ORNL	7090	F2	RS	P T=	20GRAND, 2-D FEW-GROU	40
RESONANCE INTEGRAL CALC	GA	7090	F+FAP	RSB	T=	ZUT, RESOLVED REGION	41
N RESONANCE INTEGRAL CALC	GA	7090	F+FAP	RSB	T=	TUZ, UNRESOLVED REGIO	42
ENUATION SLAB GEOMETRY	AI	7090	F2	RS	P =	GRACE1, GAMMA-RAY ATT	45
ENUATION CYL SPHERE GEOM	AI	7090	F2	RS	P =	GRACE2, GAMMA-RAY ATT	46
RATE FROM A CLOUD	AI	7090	F2	RS	P =	CLOUD, GAMMA-RAY DOSE	47
EMPERATURE VOID FRACTION	AI	7090	F2	RS	=	FUGUE, STEADY-STATE T	48
TRON SPECTRUM X-SECTIONS	AI	7090	F+FAP	RS	PLT=	TEMPEST2, THERMAL NEU	50
ECTRUM X-SECTION CALC	AI	7090	F2	RS	PLT=	FORM, FAST NEUTRON SP	51
ROXIMATION SLAB GEOMETRY	AI	7090	F2	RS	=	SAIL, 1-D 1-GP SN APP	52
1-GP S4 APPROXIMATION	AI	7090	F2	RS	=	S4 CYL CELL CODE, 1-D	53
CYCLE COST ANALYSIS	AI	7090	F2	RS	P =	AIMFIRE, URANIUM FUEL	55
CATTERING KERNEL CALC	GA	7090	F2	RS	T=	SUMMIT, CRYSTALLINE S	56
P DIFFUSION DEPLETION	AI	7090	F2	RS	LT=	SIZZLE, 1-D MULTIGROU	58
DOUBLE SN APPROX SLAB GEOM	PP	7090	F2	RS	P =	MIST, 1-D FEW-GP SN D	59
I-GP DOUBLE SN APPROX	BNWL	7090	FLOCO	RSBP	T=	GE-HAPO-S13, 1-D MULT	75
DIFFUSION CYLINDER SLAB	ORNL	7090	F2	RS	P T=	EQUIPOISE3A, 2-D 2-GP	87
GRAL X-SECTION CALC	AI	7090	F2	RS	PLT=	ARES2, RESONANCE INTE	89
OM SNAP SHIELD LEAKAGE	AI	7090	F2	RS	=	FARSE1A, DOSE RATE FR	91
SION BURNUP RZ GEOMETRY	GA	7090	F+FAP	RSBP	T=	DD8, 2-D FEW-GP DIFFU	99
TI-GP DIFFUSION PROGRAM	UK-R	7090	F+FAP	RSBP	LT=	CRAM, 1-D AND 2-D MUL	103
AND TEMPEST COMBINATION	AI	7090	F2	RS	P =	BAM, S4 CYL CELL CODE	108
S, MAXIMUM MOMENT CALC	AI	7090	F+FAP	RS	T=	4RESTRAINT PIPE STRES	109
RADIATOR FINS SNAP GEOM	AI	7090	F2	RS	=	SCARF2, SCATTER FROM	110
RING SNAP GEOMETRY	AI	7090	F2	RS	=	SCAR1, SCATTER FROM A	111

ANT DESIGN OPTIMIZATION	AI	7090	F2	RS	=	CROCK, SPACE POWER PL	112
CF MULTI-GP X-SECTIONS	LASL	7090	FLOCO	RS P	=	ZOT, GROUP-COLLAPSING	113
ANT DESIGN OPTIMIZATION	AI	7090	F2	RS	=	SHOCK, SPACE POWER PL	114
FUSION DEPLETION PROGRAM	GA	7090	F2	RSB	T=	FEVER, 1-D FEW-GP DIF	117
DIFFUSION SLAB CYL SPHERE	AI	7090	F+FAP	RS	LT=	ULCER, 1-D MULTI-GP D	118
NUM SPECTRUM X-SECTIONS	AI	7090	F+FAP	RS	LT=	QUICKIE, INFINITE MED	119
DIFFUSION SLAB CYL SPHERE	AI	7090	F+FAP	RS	LT=	FAIM, 1-D MULTI-GP DI	120
DENT KINETICS W/FEEDBACK	AI	7090	F2	RS	=	AIREK3, SPACE-INDEPEN	121
KINETICS SNAP GEOMETRY	AI	7090	F2	RS	=	SNAPKIN5/5A, 1-REGION	122
PENETRATION CALCULATION	DAC	7090	F2	RSBP	=	LIPRECAN1, MC NEUTRON	123
NICS-HYDRODYNAMICS SPHERE	LRL	7090	F2	RS P	=	CONEC, COUPLED NEUTRO	129
N APPROX SLAB CYL SPHERE	UK-W	7090	F2	RS P T=	=	W-DSN, 1-D MULTI-GP S	132
E EDIT REACTION RATES	UK-W	7090	F2	RS	=	WED, W-DSN OUTPUT TAP	133
LIDE CHAIN EQUATIONS	ORN	7090	F2	RSBP	=	NUCY, SOLUTION OF NUC	134
METERS HETEROGENEOUS CALC	FMA	7090	F2	RSBP	=	HERESY1, LATTICE PARA	136
CULATION SNAP GEOMETRY	AI	7090	F2	RS P	=	RATRAP, DOSE RATE CAL	141
ALCULATION SNAP GEOMETRY	AI	7090	F2	RS	=	MORTIMER, DOSE RATE C	142
LTIGROUP SLAB GEOMETRY	BNWL	7090	F2	RSBPLT=	=	MAC, SHIELD DESIGN MU	143
OSTS PERFORMANCE DATA	KE	7090	F2	RSBP	=	NPRFCGP, FUEL CYCLE C	146
C ELASTIC SCAT RESONANCES	AI	7090	F+FAP	RS PLT=	=	AILMOE, X-SECTION CAL	147
APPROXIMATION CYLINDER	PP	7090	F2	RS P T=	=	TOPIC, 1-D FEW-GP SN	148
SPACE CHORD DIST FUNCT	AEG	7090	F2	RS	=	DANCOFF JR, MODERATOR	150
SCRETE ORDINATE CALC	AI	7090	F2	RS PLT=	=	DTF2, 1-D MULTI-GP DI	151
RONICS-HYDRODYNAMICS CODE	AGN	7090	F2	RSB	=	HATCHET, COUPLED NEUT	153
OW EXPERIMENT ANALYSIS	AGN	7090	F2	RS P	=	CROC90, ML-1 FLUID FL	154
TI-GP DIFFUSION PROGRAM	ORN	7090	F+FAP	RS P T=	=	EXTERMINATOR, 2-D MUL	156
P TRANSFER COEFFICIENTS	LRL	7090	F+FAP	RS P T=	=	SOPHIST1/2/5, MULTI-G	160
ECAY CHAIN ANALYSIS	ORN	7090	F2	RS P	=	ISOCRUNCH, REACTION D	180
STANT CALC 0 TO 10 MEV	BNWL	7090	F+FAP	RS PLT=	=	GAMTEC2, MULTI-GP CON	185
RON PENETRATION STUDY	LASL	7090	FLOCO	RS P	=	MCS, MONTE CARLO NEUT	202
M MULTI-GP CONSTANT CALC	AGN	7090	F2	RSBPLT=	=	AGN-GAM, FAST SPECTRU	204
-D MULTI-GP DIFFUSION	WAL	7090	F+FAP	RS P T=	=	VARI-QUIR, TIME-DEP 2	212
FOR 2-D DESIGN PROGRAMS	LASL	7090	F2	RS P T=	=	DPC, DATA PREPARATION	234
CE ROUTINES ENDF TAPES	BNL	7090	F+FAP	RS P T=	=	DFSR, DATA FILE SERVI	236
MULTI-PIN FUEL ELEMENT	AGN	7090	F2	RS P	=	BOUNCE, FLUX DIST IN	237
DIFFUSION DEPLETION CODE	ORN	7090	F+FAP	RS P T=	=	ASSAULT, 2-D MULTI-GP	240
LTI-GP TRANSFER MATRICES	AGN	7090	F2	RS P T=	=	AGN-SIGMA, CALC OF MU	243
RIMENT ANALYSIS SYSTEM	BNL-SRL	7090	F2	RS PLT=	=	HAMMER, CRITICAL EXPE	277
ULATION WITH DEPLETION	WAPD	7090	F+FAP	RSBPLT=	=	LEOPARD, SPECTRA CALC	279
DISCRETE ORDINATE PROGRAM	LRC	7090	F+MAP	RS P T=	=	TDSN, 2-D MULTIGROUP	312
CED WIRING LISTS UHTREX	LASL	7090	F2	RS P	=	WIREX, COMPUTER-PRODU	315
DE X-SECTION EVALUATION	ORN	7090	F4	RSBP	=	MUFFLE, FISSIONUCL	323
S FIT SUM OF EXPONENTIALS	MIT	7090	F2	RS P	=	FRANTIC, LEAST SQUARE	324
IENCY POINT SOURCE	UM	7090	MAD	RSB	=	BURP1, DETECTOR EFFIC	164
IENCY DISK SOURCE	UM	7090	MAD	RSB	=	BURP2, DETECTOR EFFIC	165
IENCY POINT SOURCE	UM	7090	MAD	RSB	=	BURP3, DETECTOR EFFIC	166
OFRACTION SOLID CRYSTAL	UM	7090	MAD	RSB	=	BURP4, GAMMA-RAY PHOT	169
OFRACTION WELL CRYSTAL	UM	7090	MAD	RSB	=	BURP5, GAMMA-RAY PHOT	170
RS T= M0807, 2-D DIFFUSION	ABSORPTION REMOVAL	X-SECS	BAPL 6600 F4				R280
= RABBLE/WLIB/FLAT, RESONANCE	ABSORPTION, CELL	ANL 3600 F36	RSBP T				281
RS T= GADOSE/DOSET, HTGR	ACCIDENT ANALYSIS	DOSE CALC GA 1108 F4					261
M0555(ACT1), LOSS-OF-COOLANT	ACCIDENT ANALYSIS	BAPL 6600 F4	RS P T=				R284
P T= FLASH2, LOSS-OF-COOLANT	ACCIDENT ANALYSIS	BAPL 6600 F4	RS				R295
T= CONTEMPT, LOSS-OF-COOLANT	ACCIDENT ANALYSIS	PPC 7040 F+MAP	RS P				297
RATOR SPACE CHORD DIST FUNCT	AEG	7090 F2	RS	=	DANCOFF JR, MODE		150
360 F4 RS PLT= FIRE2, 1-D	AGE-DIFFUSION SLAB CYLINDER SPHERE	ROA					9
GP SN APPROXIMATION XY GEOM	AGN	7090 FLOCO	RSBP	=	2DXY, 2-D MULTI-		18
NEUTRONICS-HYDRODYNAMICS CODE	AGN	7090 F2	RSB	=	HATCHET, COUPLED		153
ID FLOW EXPERIMENT ANALYSIS	AGN	7090 F2	RS P	=	CROC90, ML-1 FLU		154
ECTRUM MULTI-GP CONSTANT CALC	AGN	7090 F2	RSBPLT=	=	AGN-GAM, FAST SP		204
T IN MULTI-PIN FUEL ELEMENT	AGN	7090 F2	RS P	=	BOUNCE, FLUX DIS		237

OF MULTI-GP TRANSFER MATRICES	AGN	7090	F2	RS P T=	AGN-SIGMA, CALC	243
CALC AGN 7090 F2 RSBPLT=	AGN-GAM, FAST SPECTRUM MULTI-GP CONSTANT					204
ICES AGN 7090 F2 RS P T=	AGN-SIGMA, CALC OF MULTI-GP TRANSFER MATR					243
IGROUP DIFFUSION DEPLETION	AI	360	F4	RSBP T=	SIZZLE, 1-D MULT	58
N CALC ELASTIC SCAT RESONANCES	AI	360	F4	RS PLT=	AILMOE, X-SECTIO	147
-GP DISCRETE ORDINATE CALC	AI	360	F4	RSBP LT=	ANISN, 1-D MULTI	151
RIC SITE REQUIREMENT STUDY	AI	360	F4	RS P T=	AISITE2, PARAMET	172
1-D MULTI-GP DIFFUSION + LIB	AI	360	F4	RSBP LT=	CAESAR4/LIBLST,	270
REDES LIBRARY UTILITY ROUTINE	AI	360	F4	RSB LT=	CLIP, FORM OR TH	271
-GP DIFFUSION DESIGN SYSTEM	AI	360	F4	RSBP T=	THREDES, 1-D FEW	273
MCNTE CARLO CELL CALCULATION	AI	360	F4	RS P T=	HWOCR-SAFE, 2-D	307
DEPENDENT REACTOR KINETICS	AI	360	F4	RS P T=	TSN, SPATIALLY-D	309
ION OF REACTOR DYNAMICS	AI	360	F+BAL	RS P T=	AIROS2A, SIMULAT	326
DIFFUSION SLAB CYLINDER SPHERE	AI	7090	F2	RS P T=	FOG, 1-D FEW-GP	28
GP DIFFUSION SLAB CYL SPHERE	AI	7090	F+FAP	RS PLT=	AIM6, 1-D MULTI-	29
BATION FOR AIM AND FOG CODES	AI	7090	F2	RS	= PERT, 1-D PERTUR	30
Y ATTENUATION SLAB GEOMETRY	AI	7090	F2	RS P	= GRACE1, GAMMA-RA	45
Y ATTENUATION CYL SPHERE GEOM	AI	7090	F2	RS P	= GRACE2, GAMMA-RA	46
DOSE RATE FROM A CLOUD	AI	7090	F2	RS P	= CLOUD, GAMMA-RAY	47
ATE TEMPERATURE VOID FRACTION	AI	7090	F2	RS	= FUGUE, STEADY-ST	48
L NEUTRON SPECTRUM X-SECTIONS	AI	7090	F+FAP	RS PLT=	TEMPEST2, THERMA	50
ON SPECTRUM X-SECTION CALC	AI	7090	F2	RS PLT=	FORM, FAST NEUTR	51
N APPROXIMATION SLAB GEOMETRY	AI	7090	F2	RS	= SAIL, 1-D 1-GP S	52
, 1-D 1-GP S4 APPROXIMATION	AI	7090	F2	RS	= S4 CYL CELL CODE	53
FUEL CYCLE COST ANALYSIS	AI	7090	F2	RS P	= AIMFIRE, URANIUM	55
IGROUP DIFFUSION DEPLETION	AI	7090	F2	RS LT=	SIZZLE, 1-D MULT	58
INTEGRAL X-SECTION CALC	AI	7090	F2	RS PLT=	ARES2, RESONANCE	89
TE FROM SNAP SHIELD LEAKAGE	AI	7090	F2	RS	= FARSE1A, DOSE RA	91
CODE AND TEMPEST COMBINATION	AI	7090	F2	RS P	= BAM, S4 CYL CELL	108
STRESS, MAXIMUM MOMENT CALC	AI	7090	F+FAP	RS T=	4RESTRAINT PIPE	109
FROM RADIATOR FINS SNAP GEOM	AI	7090	F2	RS	= SCARF2, SCATTER	110
FROM A RING SNAP GEOMETRY	AI	7090	F2	RS	= SCAR1, SCATTER F	111
ER PLANT DESIGN OPTIMIZATION	AI	7090	F2	RS	= CROCK, SPACE POW	112
ER PLANT DESIGN OPTIMIZATION	AI	7090	F2	RS	= SHOCK, SPACE POW	114
-GP DIFFUSION SLAB CYL SPHERE	AI	7090	F+FAP	RS LT=	ULCER, 1-D MULTI	118
E MEDIUM SPECTRUM X-SECTIONS	AI	7090	F+FAP	RS LT=	QUICKIE, INFINIT	119
GP DIFFUSION SLAB CYL SPHERE	AI	7090	F+FAP	RS LT=	FAIM, 1-D MULTI-	120
DEPENDENT KINETICS W/FEEDBACK	AI	7090	F2	RS	= AIREK3, SPACE-IN	121
EGION KINETICS SNAP GEOMETRY	AI	7090	F2	RS	= SNAPKIN5/5A, 1-R	122
E. CALCULATION SNAP GEOMETRY	AI	7090	F2	RS P	= RATRAP, DOSE RAT	141
ATE CALCULATION SNAP GEOMETRY	AI	7090	F2	RS	= MORTIMER, DOSE R	142
N CALC ELASTIC SCAT RESONANCES	AI	7090	F+FAP	RS PLT=	AILMOE, X-SECTIO	147
RLO SLOWING-DOWN DENSITY CALC	AI	7094	F+FAP	RS PLT=	TYCHE3, MONTE CA	149
GP DISCRETE ORDINATE CALC	AI	7090	F2	RS PLT=	DTF2, 1-D MULTI-	151
LIBRARY UTILITY PROGRAM	AI	7094	F+FAP	RS LT=	MOMUS, X-SECTION	159
RAGING X-SECTIONS PARAMETERS	AI	7094	F+FAP	RS PLT=	GRAVE, GROUP-AVE	162
DEPENDENT KINETICS W/FEEDBACK	AI	7094	F+MAP	RS P T=	AIROS, SPACE-IND	163
RIC SITE REQUIREMENT STUDY	AI	7094	F+FAP	RS P T=	AISITE2, PARAMET	172
ERHEAD, EXTERNAL+INTERNAL DOSE	AI	7094	F+FAP	RS PLT=	CURIE/DOSE/THUND	196
GHT OPTIMIZATION DOSE CALC	AI	7094	F2	RS P	= SHOE, SHIELD WEI	197
ENT STEADY-STATE HEAT TRANSFER	AI	7094	F2	RS P T=	HEATING2, TRANSI	198
INTEGRAL X-SECTION CALC	AI	7094	F+MAP	RS PLT=	TRIX1, RESONANCE	208
DEF CALC FOR ANGULAR DIST DATA	AI	7094	F+MAP	RS P	= CHAD, LEGENDRE C	215
DYNAMIC CYCLE ANALYSIS	AI	7094	F2	RS PLT=	CYCLOPS1, THERMO	244
I-CHANNEL 2-D 2-PHASE FLOW	AI	7094	F2	RS P	= FLOW-MODEL, MULT	246
ICAL OPTICAL MODEL X-SECTIONS	AI	7094	F2	RS P T=	2PLUS, NON-SPHER	254
ANSIENT FLOW AND HEAT TRANSFER	AI	7094	F2	RS P T=	TRANS-FUGUE1, TR	268
DEPENDENT REACTOR KINETICS	AI	7094	F4	RS P T=	TSN, SPATIALLY-D	309
ANCES AI 7090 F+FAP RS PLT=	AILMOE, X-SECTION CALC ELASTIC SCAT RESON					147
ANCES AI 360 F4 RS PLT=	AILMOE, X-SECTION CALC ELASTIC SCAT RESON					147
= PERT, 1-D PERTURBATION FOR AIM AND FOG CODES	AI	7090	F2	RS		30

= PERT, 1-D PERTURBATION FOR	AIM AND FOG CODES	CDC 1604 F63	RS	30
= PERT, 1-D PERTURBATION FOR	AIM AND FOG CODES	BOEC 360 F4	RS P	30
ERE AI 7090 F+FAP RS PLT=	AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPH			29
ERE CDC 1604 F63 RS PLT=	AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPH			29
AI 7090 F2 RS P =	AIMFIRE, URANIUM FUEL CYCLE COST ANALYSIS			55
BACK AI 7090 F2 RS =	AIREK3, SPACE-INDEPENDENT KINETICS W/FEED			121
BACK ROA 360 F4 RS P =	AIREK3, SPACE-INDEPENDENT KINETICS W/FEED			121
AI 360 F+BAL RS P T=	AIROS2A, SIMULATION OF REACTOR DYNAMICS			326
ACK AI 7094 F+MAP RS P T=	AIROS, SPACE-INDEPENDENT KINETICS W/FEEDB			163
Y AI 7094 F+FAP RS P T=	AISITE2, PARAMETRIC SITE REQUIREMENT STUD			172
Y AI 360 F4 RS P T=	AISITE2, PARAMETRIC SITE REQUIREMENT STUD			172
RS P T= TOAD, PROCESSING OF	ANALYZER GAMMA-RAY SPECTRA GGA 1108 F4			333
CHAD, LEGENDRE COEF CALC FOR	ANGULAR DIST DATA AI 7094 F+MAP RS P =			215
GCOEF3, LEGENDRE COEF CALC FOR	ANGULAR DIST GA 7044 F4 RS P =	LE		217
C AI 360 F4 RSBPLT=	ANISN, 1-D MULTI-GP DISCRETE ORDINATE CAL			151
TOR EXCURSION CALCULATIONS	ANL 360 F4 RS P T=	FORE2, FAST REAC		174
GP DISCRETE ORDINATE PROGRAM	ANL 360 F4 RS P T=	DTF4, 1-D MULTI-		209
ON SPECTRUM X-SECTION CALC	ANL 3600 F63 RS LT=	GAM1, FAST NEUTR		33
, TRANSFER FUNCTION SYNTHESIS	ANL 3600 F63 RSBP T=	TRAFICORPORATION		135
GN MULTIGROUP SLAB GEOMETRY	ANL 3600 F63 RSBPLT=	MAC, SHIELD DESI		143
TEMPERATURE CALC CYLINDER	ANL 3600 F63 RSBP T=	ARGUS, TRANSIENT		152
PENDENT INVERSE KINETICS CALC	ANL 3600 F63 RSBP =	R102, SPACE-INDE		168
D NUCLEUS X-SECTION CALC	ANL 3600 F63 RSBP =	NEARREX, COMPOUN		171
SITE REQUIREMENT STUDY	ANL 3600 F63 RS P T=	R153, PARAMETRIC		172
N LIBRARY UTILITY ROUTINE	ANL 3600 F63 RSBP T=	XLBIT, X-SECTIO		181
ERMAL SPECTRUM X-SECTION CALC	ANL 3600 F63 RS P T=	THERMOS(ANL), TH		184
ANCE INTEGRAL CALC 2-REG CELL	ANL 3600 F63 RSBP T=	RIFF-RAFF, RESON		213
ANCE INTEGRAL CALC HOMOGENEOUS	ANL 3600 F63 RSBP =	MISH-MASH, RESON		214
CE COUNTING DATA REDUCTION	ANL 3600 F63 RSBP =	COINC, COINCIDEN		248
ICAL OPTICAL MODEL X-SECTIONS	ANL 3600 F63 RSBP T=	2PLUS, NON-SPHER		254
PENDENT KINETICS KEX OPTIONS	ANL 3600 F63 RSBP =	R101, SPACE-INDE		255
-GP DIFFUSION SLAB CYL SPHERE	ANL 3600 F63 RS P T=	MACH1, 1-D MULTI		262
, RESONANCE ABSORPTION, CELL	ANL 3600 F36 RSBP T=	RABBLE/WLIB/FLAT		281
LTI-GP DISCRETE ORDINATE CALC	ANL 3600 F36 RS P T=	SNARG-1D, 1-D MU		288
EC 625 F4 RS P =	AVOID, ANNULAR VOID X-SECTION CALCULATION	G		276
COUPLED NEUTRON-HYDRODYNAMICS	APCA 7094 F2 RS P =	WEAK EXPLOSION,		145
ION CALCULATION R-Z GEOMETRY	APCA 7094 F4 RS P =	MARS, 2-D EXCURS		293
AI 7090 F2 RS PLT=	ARES2, RESONANCE INTEGRAL X-SECTION CALC			89
CDC 1604 F63 RS PLT=	ARES2, RESONANCE INTEGRAL X-SECTION CALC			89
R ANL 3600 F63 RSBP T=	ARGUS, TRANSIENT TEMPERATURE CALC CYLINDE			152
CODE ORNL 7090 F+FAP RS P T=	ASSAULT, 2-D MULTI-GP DIFFUSION DEPLETION			240
PW 1604 F+CDP RS P T=	LAG, ASSEMBLER FOR FLOCO2 INSTRUCTION SET			186
RS P = GRACE1, GAMMA-RAY	ATTENUATION SLAB GEOMETRY	AI 7090 F2		45
3 RS P = GRACE1, GAMMA-RAY	ATTENUATION SLAB GEOMETRY	CDC 1604 F6		45
RS P = GRACE2, GAMMA-RAY	ATTENUATION CYL SPHERE GEOM	AI 7090 F2		46
3 RS P = GRACE2, GAMMA-RAY	ATTENUATION CYL SPHERE GEOM	CDC 1604 F6		46
GEC 625 F4 RS P =	AVOID, ANNULAR VOID X-SECTION CALCULATION			276
ERE CDC 3600 F63 RSBP =	AX1, COUPLED NEUTRONICS-HYDRODYNAMICS SPH			102
DLES LASL 7094 F2 RS P =	AXFLU, HEAT TRANSFER MOLTEN FUEL TUBE BUN			182
T= SAFE-PCRS, STRESS ANALYSIS	AXISYMMETRIC LOAD	GA 7044 F4 RS P		250
SAFE-AXISYM, STRESS ANALYSIS	AXISYMMETRIC LOAD	GA 7044 F4 RS P T=		251
LSHELL2, SHELL STRESS ANALYSIS	AXISYM LOAD	BAPL 6600 F4 RS P T=	SEA	R282
SPH PW 1604 F63 RS P =	AX-TNT, COUPLED NEUTRONICS-HYDRODYNAMICS			191
DLES LASL 7094 F2 RS P =	AXTHRM, HEAT TRANSFER SOLID FUEL TUBE BUN			183
TION AI 7090 F2 RS P =	BAM, S4 CYL CELL CODE AND TEMPEST COMBINA			108
LES 14.5-14,500PSIA 32-472DEGF	BAPL 2000 F4 RS =	WATER, STEAM TAB		R267
NALYSIS CYL FUEL ELEMENT	BAPL 6600 F4 RS T=	CYGR02, STRESS A		R266
SWELLING TEMPERATURE STUDY	BAPL 6600 F4 RS P T=	FIGRO, LSBR FUEL		R272
SPACE-TIME DIFFUSION 3-GEOM	BAPL 6600 F4 RS P T=	WIGL2, 1-D 2-GP		R274
FEW-GP DIFFUSION DEPLETION	BAPL 6600 F+ASC RS T=	PDQ7, 1,2 OR 3-D		R275
SION ABSORPTION REMOVAL X-SECS	BAPL 6600 F4 RS T=	M0807, 2-D DIFFU		R280

L STRESS ANALYSIS AXISYM LOAD	BAPL 6600 F4	RS P T=	SEALSHL2, SHL	R282
ANALYSIS LINEAR ELASTIC SYSTEMS	BAPL 6600 F4	RS P T=	M0552, DYNAMIC A	R283
S-OF-COOLANT ACCIDENT ANALYSIS	BAPL 6600 F4	RS P T=	M0555(ACT1), LOS	R284
, RESONANCE INTEGRAL HEX CELL	BAPL 6600 F4	RS P T=	RESQ2/RESQ0/DBF1	R285
ENT HEAT CONDUCTION PROGRAM	BAPL 6600 F4	RS P T=	HOT2, 2-D TRANSI	R286
TABLES 14.5-2538 PSIA	BAPL 6600 F4	RS P T=	M0899/HOH, STEAM	R294
COOLANT ACCIDENT ANALYSIS	BAPL 6600 F4	RS P T=	FLASH2, LOSS-OF-	R295
JOINT DEPLETION FISSION PRODUCT	BAPL 6600 F4	RS P =	CINDER(M0102), P	313
ARES FIT FISSION NUCLEIDE DATA	BAPL 6600 F4	RS P T=	DAFT1, LEAST SQU	R327
STIC STRESS OF PIPING SYSTEM	BAPL 6600 F4	RS P T=	M0457(PIPE), ELA	R329
R FLOW TRANSIENT ANALYSIS	BAPL 6600 F4	RS P T=	FLOT1(M0219), PW	R331
GP DISCRETE ORDINATE PROGRAM	BEC 625 F4	RS P T=	DTF4, 1-D MULTI-	209
ALC GESJ 635 F4 RS PLT=	BISYN, 2-D MULTI-GP DIFFUSION SYNTHESIS C			287
4 RS P = COMBCO, COMBINED	B-W MULTI-LEVEL CONVOLUTION PP 7040 F			203
P = FASDOP, X-SECTIONS FROM	B-W RESONANCE DATA GA 7044 F4 RS			216
RS P = EXT, X-SECTIONS FROM	B-W RESONANCE PARAMETERS WAL 7094 F2			238
NSFER GA 7044 F4 RS T=	BLOOST5, POINT-KINETICS WITH 2-D HEAT TRA			205
FER GGA 1108 F4 RS P T=	BLCOST6, COMBINED KINETICS 2-D HEAT TRANS'			303
KE 7094 F2 RSBP T= PTH1,	BLOWDOWN PRESSURE TEMPERATURE HISTORY			155
RSBP T= SATURATED BLOWDOWN1,	BLOWDOWN ANALYSIS LOFT KE 7094 F+MAP			200
7094 F+MAP RSBP T= SATURATED	BLOWDOWN1, BLOWDOWN ANALYSIS LOFT KE			200
RS P = WATER-HAMMER, LIQUID	BLOWDOWN ANALYSIS LOFT UG 360 F4			278
-OF-COOLANT THERMAL ANALYSIS	BMI 6400 F4 RS P T=	NURLOC-1.0, LOSS		328
COOLANT + EMERGENCY COOLING	BMI 6400 F4 RS P T=	ECCSA1, LOSS-OF-		330
SERVICE ROUTINES ENDF TAPES	BNL 7090 F+FAP RS P T=	DFSR, DATA FILE		236
CAL EXPERIMENT ANALYSIS SYSTEM	BNL-SRL 7090 F2 RS PLT=	HAMMER, CRITI		277
CAL EXPERIMENT ANALYSIS SYSTEM	BNL-SRL 360 F4 RS PLT=	HAMMER, CRITI		277
P CONSTANT CALC 0 TO 10 MEV	BNWL 1107 F4 RS PLT=	GAMTEC2, MULTI-G		185
P DIFFUSION SLAB CYL SPHERE	BNWL 1107 F4 RS P T=	HFN, 1-D MULTI-G		241
RBATION XY RZ RTHETA GEOMETRY	BNWL 1108 F4 RS P =	PERT4, 2-D PERTU		304
L MODE FAST REACTOR X-SEC CALC	BNWL 1108 F4 RS P T=	FCC4, FUNDAMENTA		306
ROUP DIFFUSION AND DEPLETION	BNWL 1108 F4 RS P T=	2DB, 2-D MULTIGR		325
MULTI-GP DOUBLE SN APPROX	BNWL 7090 FLOCO RSBP T=	GE-HAPO-S13, 1-D		75
GN MULTIGROUP SLAB GEOMETRY	BNWL 7090 F2 RSBPLT=	MAC, SHIELD DESI		143
P CONSTANT CALC 0 TO 10 MEV	BNWL 7090 F+FAP RS PLT=	GAMTEC2, MULTI-G		185
BATION FOR AIM AND FOG CODES	BOEC 360 F4 RS P =	PERT, 1-D PERTUR		30
L NEUTRON SPECTRUM X-SECTIONS	BOEC 360 F4 RS PLT=	TEMPEST2, THERMA		50
I-GP DIFFUSION SLAB CYL SPHERE	BOEC 360 F4 RS P T=	FAIMOS, 1-D MULT		120
NT AGN 7090 F2 RS P =	BOUNCE, FLUX DIST IN MULTI-PIN FUEL ELEME			237
HEAT TRANSFER MOLTEN FUEL TUBE	BUNDLES LASL 7094 F2 RS P =	AXFLU,		182
HEAT TRANSFER SOLID FUEL TUBE	BUNDLES LASL 7094 F2 RS P =	AXTHRM,		183
T= DDB, 2-D FEW-GP DIFFUSION	BURNUP RZ GEOMETRY GA 7090 F+FAP RSBP			99
LT= LASER, SPECTRUM CALC WITH	BURNUP CYL LATTICE WAPD 7094 F+MAP RS P			249
BURNUP, HEAVY ELEMENT ISOTOPIC	BURNUP ANALYSIS GEV 635 F4 RS P T=			311
LYSIS GEV 635 F4 RS P T=	BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANA			311
1108 F4 RS P T=	BURNUP POWER DISTRIBUTION SEARCH GGA			319
UM 7090 MAD RSB =	BURP1, DETECTOR EFFICIENCY POINT SOURCE			164
UM 7090 MAD RSB =	BURP2, DETECTOR EFFICIENCY DISK SOURCE			165
UM 7090 MAD RSB =	BURP3, DETECTOR EFFICIENCY POINT SOURCE			166
TAL UM 7090 MAD RSB =	BURP4, GAMMA-RAY PHOTOFRACITION SOLID CRYST			169
AL UM 7090 MAD RSB =	BURP5, GAMMA-RAY PHOTOFRACITION WELL CRYST			170
RESONANCE INTEGRAL CALCULATION	BW 2000 F4 RS =	STRIP, RESOLVED		305
LIB AI 360 F4 RSBPLT=	CAESAR4/LIBLST, 1-D MULTI-GP DIFFUSION +			270
AT, VIEW FACTOR SHIELDING CODE	CAVITY GEOM GA 1108 F4 RS P T=	MUSC		259
FESHBACH INELASTIC SCATTERING	CDC 1604 F63 RS P =	HAFEVER, HAUSER-		14
DIFFUSION SLAB CYLINDER SPHERE	CDC 1604 F63 RS P T=	FOG, 1-D FEW-GP		28
GP DIFFUSION SLAB CYL SPHERE	CDC 1604 F63 RS PLT=	AIM6, 1-D MULTI-		29
BATION FOR AIM AND FOG CODES	CDC 1604 F63 RS =	PERT, 1-D PERTUR		30
-GROUP DIFFUSION XYZ GEOMETRY	CDC 1604 F63 RS P T=	WHIRLAWAY, 3-D 2		32
ON SPECTRUM X-SECTION CALC	CDC 1604 F63 RS PLT=	GAM1, FAST NEUTR		33
-GROUP DIFFUSION SLAB CYLINDER	CDC 1604 F63 RS P T=	20GRAND, 2-D FEW		40

Y ATTENUATION SLAB GEOMETRY	CDC 1604 F63	RS P = GRACE1, GAMMA-RA	45
Y ATTENUATION CYL SPHERE GEOM	CDC 1604 F63	RS P = GRACE2, GAMMA-RA	46
DOSE RATE FROM A CLOUD	CDC 1604 F63	RS P = CLOUD, GAMMA-RAY	47
L NEUTRON SPECTRUM X-SECTIONS	CDC 1604 F63	RS PLT= TEMPEST2, THERMA	50
ON SPECTRUM X-SECTION CALC	CDC 1604 F63	RS PLT= FORM, FAST NEUTR	51
, 1-C 1-GP S4 APPROXIMATION	CDC 1604 F63	RS = S4 CYL CELL CODE	53
IGROUP DIFFUSION DEPLETION	CDC 1604 F63	RS LT= SIZZLE, 1-D MULT	58
2-GP DIFFUSION CYLINDER SLAB	CDC 1604 F63	RS P T= EQUIPOISE3A, 2-D	87
INTEGRAL X-SECTION CALC	CDC 1604 F63	RS PLT= ARES2, RESONANCE	89
GP DIFFUSION SLAB CYL SPHERE	CDC 1604 F63	RS PLT= FAIM, 1-D MULTI-	120
TRONICS-HYDRODYNAMICS SPHERE	CDC 3600 F63	RSBP = AX1, COUPLED NEU	102
IVITY AND POWER DISTRIBUTION	CDC 3600 F63	RSBP T= FLARE, 3-D REACT	167
RESCNANCE INTEGRAL CALC 2-REG	CELL ANL 3600 F63	RSBP T= RIFF-RAFF,	213
IB/FLAT, RESONANCE ABSORPTION,	CELL ANL 3600 F36	RSBP T= RABBLE/WL	281
O/DBF1, RESONANCE INTEGRAL HEX	CELL BAPL 6600 F4	RS P T= RESQ2/RESQ	R285
I 7090 F2 RS = S4 CYL	CELL CODE, 1-D 1-GP S4 APPROXIMATION	A	53
DC 1604 F63 RS = S4 CYL	CELL CODE, 1-D 1-GP S4 APPROXIMATION	C	53
O90 F2 RS P = BAM, S4 CYL	CELL CODE AND TEMPEST COMBINATION	AI 7	108
= HWOGR-SAFE, 2-D MONTE CARLO	CELL CALCULATION	AI 360 F4 RS P T	307
PW 1604 F63 RS P T= PIP,	CENTRIFUGAL PUMP IMPELLER DESIGN STUDY		187
DATA AI 7094 F+MAP RS P =	CHAD, LEGENDRE COEF CALC FOR ANGULAR DIST		215
= NUCY, SOLUTION OF NUCLIDE	CHAIN EQUATIONS	ORNL 7090 F2 RSBP	134
= NUCY, SOLUTION OF NUCLIDE	CHAIN EQUATIONS	ORNL 1604 F63 RSBP	134
= ISOCRUNCH, REACTION DECAY	CHAIN ANALYSIS	ORNL 7090 F2 RS P	180
= ISOCRUNCH, REACTION DECAY	CHAIN ANALYSIS	ORNL 1604 F63 RS P	180
= DANCOFF JR, MODERATOR SPACE	CHORD DIST FUNCT	AEG 7090 F2 RS	150
ODUCT BAPL 6600 F4 RS P =	CINDER(M0102), POINT DEPLETION FISSION PR		313
TINE AI 360 F4 RSB LT=	CLIP, FORM OR THREDES LIBRARY UTILITY ROU		271
UD, GAMMA-RAY DOSE RATE FROM A	CLOUD	AI 7090 F2 RS P = CLO	47
AI 7090 F2 RS P =	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD		47
CDC 1604 F63 RS P =	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD		47
SRL 360 F4 RS P =	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD		47
PW 1604 F63 RS P =	CMPXMAT, TRANSFER FUNCTION EVALUATION		188
NL 3600 F63 RSBP = COINC,	COINCIDENCE COUNTING DATA REDUCTION	A	248
N ANL 3600 F63 RSBP =	COINC, COINCIDENCE COUNTING DATA REDUCTIO		248
ION PP 7040 F4 RS P =	COMBCO, COMBINED B-W MULTI-LEVEL CONVOLUT		203
08 F4 RS P T= SAFE-3D, 3-D	COMPOSITE STRUCTURE STRESS STUDY	GGA 11	332
3600 F63 RSBP = NEARREX,	COMPOUND NUCLEUS X-SECTION CALC	ANL	171
ASL 7090 F2 RS P = WIREX,	COMPUTER-PRODUCED WIRING LISTS UHTREX	L	315
E-CREEP, VISCOELASTIC ANALYSIS	CONCRETE	GGA 1108 F4 RS P = SAF	300
SPHERE LRLL 7090 F2 RS P =	CONEC, COUPLED NEUTRONICS-HYDRODYNAMICS S		129
7044 F4 RS = CORE, CORE	CONFIGURATION FUEL TEMPERATURE CODE GA		233
44 F4 RS T= PRECON, HTGR	CONTAINMENT PRESSURE POST RUPTURE GA	70	228
IS PPC 7040 F+MAP RS P T=	CONTEMPT, LOSS-OF-COOLANT ACCIDENT ANALYS		297
MBCO, COMBINED B-W MULTI-LEVEL	CONVOLUTION	PP 7040 F4 RS P = CO	203
1, LOSS-OF-COOLANT + EMERGENCY	COOLING	BMI 6400 F4 RS P T= ECCSA	330
RS P = GAZELLE5, GAS-COOLED	CORE THERMAL DESIGN STUDY	GA 7044 F4	232
CODE GA 7044 F4 RS =	CORE, CORE CONFIGURATION FUEL TEMPERATURE		233
= AIMFIRE, URANIUM FUEL CYCLE	COST ANALYSIS	AI 7090 F2 RS P	55
RSBP = NPRFCCP, FUEL CYCLE	COSTS PERFORMANCE DATA	KE 7090 F2	146
RS P = WAMPUM, FUEL CYCLE	COSTS PERFORMANCE STUDY	GA 7044 F4	224
TEAM GENERATOR DESIGN CRITERIA	COSTS	GA 7044 F4 RS P T= STMGEN, S	227
RSBP = CCINC, COINCIDENCE	COUNTING DATA REDUCTION	ANL 3600 F63	248
CDC 3600 F63 RSBP = AX1,	COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE		102
RLL 7090 F2 RS P = CONEC,	COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE L		129
F2 RS P = WEAK EXPLOSION,	COUPLED NEUTRON-HYDRODYNAMICS	APDA 7094	145
7090 F2 RSB = HATCHET,	COUPLED NEUTRONICS-HYDRODYNAMICS CODE AGN		153
1604 F63 RS P = AX-TNT,	COUPLED NEUTRONICS-HYDRODYNAMICS SPH	PW	191
1604 F63 RS P T= JUPITOR1,	COUPLED-CHANNEL X-SEC EVALUATION	ORNL	308
APES WAL 7094 F2 RS T=	CPS, SC4020 PLOTS FROM SCISRS X-SECTION T		239
RAM UK-R 7090 F+FAP RSBPLT=	CRAM, 1-D AND 2-D MULTI-GP DIFFUSION PROG		103

RL 7090 F2	RS PLT= HAMMER,	CRITICAL EXPERIMENT ANALYSIS SYSTEM BNL-S	277
RL 360 F4	RS PLT= HAMMER,	CRITICAL EXPERIMENT ANALYSIS SYSTEM BNL-S	277
IS AGN 7090 F2	RS P =	CROC90, ML-1 FLUID FLOW EXPERIMENT ANALYS	154
ION AI 7090 F2	RS =	CROCK, SPACE POWER PLANT DESIGN OPTIMIZAT	112
PES GA 7044 F4	RSBP T=	CROSSPLOT, SC4020 PLOTS FROM X-SECTION TA	207
7090 F2	RS T= SUMMIT,	CRYSTALLINE SCATTERING KERNEL CALC GA	56
GAMMA-RAY PHOTOFRACTION SOLID	CRYSTAL UM 7090 MAD	RSB = BURP4,	169
, GAMMA-RAY PHOTOFRACTION WELL	CRYSTAL UM 7090 MAD	RSB = BURP5	170
ON PW 1604 F63	RS =	CSP1, SN X-SECTION LIBRARY TAPE PREPARATI	194
ION PW 1604 F63	RS =	CSP2A, SN X-SECTION LIBRARY TAPE PREPARAT	193
OINTS KAPL 6600 F+ASC	RS P T=	CURFIT, CURVE FITTING EXPERIMENTAL DATA P	R 43
DOSE AI 7094 F+FAP	RS PLT=	CURIE/DOSE/THUNDERHEAD, EXTERNAL+INTERNAL	196
PL 6600 F+ASC	RS P T=	CURFIT, CURVE FITTING EXPERIMENTAL DATA POINTS KA	R 43
= LASL LEAST SQUARES, GENERAL	CURVE FITTING	LASL 7094 F4 RS	62
EAST SQUARES EXPONENTIAL DECAY	CURVES LRL 7094 F2	RS P = EXPALS, L	321
AI 7094 F2	RS PLT=	CYCLOPS1, THERMODYNAMIC CYCLE ANALYSIS	244
BAPL 6600 F4	RS T=	CYGR02, STRESS ANALYSIS CYL FUEL ELEMENT	R266
E3, 2-D 2-GROUP DIFFUSION SLAB	CYL ORNL 7090 F2	RS P T= EQUIPOIS	39
AI 7090 F2	RS = S4	CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION	53
CDC 1604 F63	RS = S4	CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION	53
7090 F2	RS P = BAM, S4	CYL CELL CODE AND TEMPEST COMBINATION AI	108
T= CYGR02, STRESS ANALYSIS	CYL FUEL ELEMENT	BAPL 6600 F4 RS	R266
SER, SPECTRUM CALC WITH BURNUP	CYL LATTICE WAPD 7094 F+MAP	RS PLT= LA	249
ULTI-GP FLUX SYNTHESIS PROGRAM	CYL PW 1604 F63	RS T= 2DXYL, 3-D M	192
6, 1-D MULTI-GP DIFFUSION SLAB	CYL SPHERE AI 7090 F+FAP	RS PLT= AIM	29
6, 1-D MULTI-GP DIFFUSION SLAB	CYL SPHERE CDC 1604 F63	RS PLT= AIM	29
GRACE2, GAMMA-RAY ATTENUATION	CYL SPHERE GEOM AI 7090 F2	RS P =	46
GRACE2, GAMMA-RAY ATTENUATION	CYL SPHERE GEOM CDC 1604 F63	RS P =	46
R, 1-D MULTI-GP DIFFUSION SLAB	CYL SPHERE AI 7090 F+FAP	RS LT= ULCE	118
M, 1-D MULTI-GP DIFFUSION SLAB	CYL SPHERE AI 7090 F+FAP	RS LT= FAI	120
M, 1-D MULTI-GP DIFFUSION SLAB	CYL SPHERE CDC 1604 F63	RS PLT= FAI	120
S, 1-D MULTI-GP DIFFUSION SLAB	CYL SPHERE BOEC 360 F4	RS P T= FAIMO	120
N, 1-D MULTI-GP SN APPROX SLAB	CYL SPHERE UK-W 7090 F2	RS P T= W-DS	132
N, 1-D MULTI-GP SN APPROX SLAB	CYL SPHERE PW 1604 LAG1	RS P = MGDS	211
N, 1-D MULTI-GP DIFFUSION SLAB	CYL SPHERE BNWL 1107 F4	RS P T= HF	241
1, 1-D MULTI-GP DIFFUSION SLAB	CYL SPHERE ANL 3600 F63	RS P T= MACH	262
FIRE2, 1-D AGE-DIFFUSION SLAB	CYLINDER SPHERE ROA 360 F4	RS PLT=	9
FOG, 1-D FEW-GP DIFFUSION SLAB	CYLINDER SPHERE AI 7090 F2	RS P T=	28
FCG, 1-D FEW-GP DIFFUSION SLAB	CYLINDER SPHERE CDC 1604 F63	RS P T=	28
, 2-D FEW-GROUP DIFFUSION SLAB	CYLINDER ORNL 7090 F2	RS P T= 20GRAND	40
, 2-D FEW-GROUP DIFFUSION SLAB	CYLINDER CDC 1604 F63	RS P T= 20GRAND	40
QUIPCISE3A, 2-D 2-GP DIFFUSION	CYLINDER SLAB ORNL 7090 F2	RS P T= E	87
QUIPCISE3A, 2-D 2-GP DIFFUSION	CYLINDER SLAB CDC 1604 F63	RS P T= E	87
C, 1-D FEW-GP SN APPROXIMATION	CYLINDER PP 7040 F4	RS P T= TOPI	148
C, 1-D FEW-GP SN APPROXIMATION	CYLINDER PP 7090 F2	RS P T= TOPI	148
US, TRANSIENT TEMPERATURE CALC	CYLINDER ANL 3600 F63	RSBP T= ARG	152
UTRON PENETRATION CALCULATION	DAC 7090 F2	RSBP = LIPRECAN1, MC NE	123
DATA BAPL 6600 F4	RS P T=	DAFT1, LEAST SQUARES FIT FISSILE NUCLIDE	R327
= CCINC, COINCIDENCE COUNTING	DATA REDUCTION ANL 3600 F63	RSBP	248
NCT AEG 7090 F2	RS =	DANCOFF JR, MODERATOR SPACE CHORD DIST FU	150
X-SECTIONS FROM B-W RESONANCE	DATA GA 7044 F4	RS P = FASDOP,	216
LYSIS OF PULSED NEUTRON SOURCE	DATA GESJ 635 F4	RS P T= EXPN, ANA	258
ST SQUARES FIT FISSILE NUCLIDE	DATA BAPL 6600 F4	RS P T= DAFT1, LEA	R327
DRE COEF CALC FOR ANGULAR DIST	DATA AI 7094 F+MAP	RS P = CHAD, LEGEN	215
BNL 7090 F+FAP	RS P T= DFSR,	DATA FILE SERVICE ROUTINES ENDF TAPES	236
IT, CURVE FITTING EXPERIMENTAL	DATA POINTS KAPL 6600 F+ASC	RS P T= CURF	R 43
LASL 7090 F2	RS P T= DPC,	DATA PREPARATION FOR 2-D DESIGN PROGRAMS	234
TRY GA 7090 F+FAP	RSBP T=	DOB, 2-D FEW-GP DIFFUSION BURNUP RZ GEOME	99
MAXIMUM YIELD FROM REACTION OR	DECAY ORNL 1604 F63	RS P = ISOTOPES,	179
RS P = ISOCRUNCH, REACTION	DECAY CHAIN ANALYSIS	ORNL 7090 F2	180
RS P = ISCCRUNCH, REACTION	DECAY CHAIN ANALYSIS	ORNL 1604 F63	180

ALS, LEAST SQUARES EXPONENTIAL	DECAY CURVES	LRL	7094 F2	RS P = EXP	321
ZZLE, 1-D MULTIGROUP DIFFUSION	DEPLETION	AI	7090 F2	RS LT= SI	58
ZZLE, 1-D MULTIGROUP DIFFUSION	DEPLETION	CDC	1604 F63	RS LT= SI	58
ZZLE, 1-D MULTIGROUP DIFFUSION	DEPLETION	AI	360 F4	RSBP T= SI	58
= FEVER, 1-D FEW-GP DIFFUSION	DEPLETION	PROGRAM	GA	7090 F2	RSB T
AD-FEVER, 1-D FEW-GP DIFFUSION	DEPLETION	GA	7044 F4	RS P T= RELO	221
SSAULT, 2-D MULTI-GP DIFFUSION	DEPLETION	CODE	ORNL	7090 F+FAP	RS P T= A
F-BURN, 1-D MULTI-GP DTF4 WITH	DEPLETION	LASL	7030 F4	RS P T= DT	269
7, 1,2 OR 3-D FEW-GP DIFFUSION	DEPLETION	BAPL	6600 F+ASC	RS T= PDQ	R275
PARD, SPECTRA CALCULATION WITH	DEPLETION	WAPD	7090 F+FAP	RSBP T= LEO	279
RS P = CINDER(M0102), POINT	DEPLETION	FISSION	PRODUCT	BAPL	6600 F4
, 1-D MULTIGROUP DIFFUSION AND	DEPLETION	GGA	1108 F4	RS P T= FEVER7	318
, 2-D MULTIGROUP DIFFUSION AND	DEPLETION	BNWL	1108 F4	RS P T= 2DB	325
5, 2-D FEW-GROUP DIFFUSION AND	DEPLETION	IBM	360 F+BAL	RS P T= PDQ	R336
= CROCK, SPACE POWER PLANT	DESIGN OPTIMIZATION	AI	7090 F2	RS	112
= SHOCK, SPACE POWER PLANT	DESIGN OPTIMIZATION	AI	7090 F2	RS	114
090 F2	RSBP T= MAC, SHIELD	DESIGN	MULTIGROUP	SLAB GEOMETRY	BNWL 7
600 F63	RSBP T= MAC, SHIELD	DESIGN	MULTIGROUP	SLAB GEOMETRY	ANL 3
PIP, CENTRIFUGAL PUMP IMPELLER	DESIGN STUDY	PW	1604 F63	RS P T=	187
P T= STMGEN, STEAM GENERATOR	DESIGN CRITERIA	COSTS	GA	7044 F4	RS
ELLE5, GAS-COOLED CORE THERMAL	DESIGN STUDY	GA	7044 F4	RS P = GAZ	232
DPC, DATA PREPARATION FOR 2-D	DESIGN PROGRAMS	LASL	7090 F2	RS P T=	234
THREDES, 1-D FEW-GP DIFFUSION	DESIGN SYSTEM	AI	360 F4	RSBP T=	273
M 7090 MAD	RSB = BURP1,	DETECTOR	EFFICIENCY	POINT SOURCE	U
M 7090 MAD	RSB = BURP2,	DETECTOR	EFFICIENCY	DISK SOURCE	U
M 7090 MAD	RSB = BURP3,	DETECTOR	EFFICIENCY	POINT SOURCE	U
ES	BNL	7090 F+FAP	RS P T=	DFSR, DATA FILE SERVICE	ROUTINES ENDF TAP
F2	RS P T= FOG, 1-D FEW-GP	DIFFUSION	SLAB CYLINDER	SPHERE	AI
F63	RS P T= FOG, 1-D FEW-GP	DIFFUSION	SLAB CYLINDER	SPHERE	CDC
AP	RS PLT= AIM6, 1-D MULTI-GP	DIFFUSION	SLAB CYL	SPHERE	AI
RS	PLT= AIM6, 1-D MULTI-GP	DIFFUSION	SLAB CYL	SPHERE	CDC
S	P T= WHIRLAWAY, 3-D 2-GROUP	DIFFUSION	XYZ GEOMETRY	ORNL	7090 F2
S	P T= WHIRLAWAY, 3-D 2-GROUP	DIFFUSION	XYZ GEOMETRY	CDC	1604 F63
P	T= EQUIPOISE3, 2-D 2-GROUP	DIFFUSION	SLAB CYL	ORNL	7090 F2
S	P T= 20GRAND, 2-D FEW-GROUP	DIFFUSION	SLAB CYLINDER	ORNL	7090 F2
S	P T= 20GRAND, 2-D FEW-GROUP	DIFFUSION	SLAB CYLINDER	CDC	1604 F63
S	LT= SIZZLE, 1-D MULTIGROUP	DIFFUSION	DEPLETION	AI	7090 F2
S	LT= SIZZLE, 1-D MULTIGROUP	DIFFUSION	DEPLETION	CDC	1604 F63
SBP	T= SIZZLE, 1-D MULTIGROUP	DIFFUSION	DEPLETION	AI	360 F4
RS	P T= EQUIPOISE3A, 2-D 2-GP	DIFFUSION	CYLINDER	SLAB	ORNL
RS	P T= EQUIPOISE3A, 2-D 2-GP	DIFFUSION	CYLINDER	SLAB	CDC
F+FAP	RSBP T= DDB, 2-D FEW-GP	DIFFUSION	BURNUP	RZ GEOMETRY	GA
T= CRAM, 1-D AND 2-D MULTI-GP	DIFFUSION	PROGRAM	UK-R	7090 F+FAP	RSBP
RSB	T= FEVER, 1-D FEW-GP	DIFFUSION	DEPLETION	PROGRAM	GA
P	RS LT= ULCER, 1-D MULTI-GP	DIFFUSION	SLAB CYL	SPHERE	AI
AP	RS LT= FAIM, 1-D MULTI-GP	DIFFUSION	SLAB CYL	SPHERE	AI
RS	PLT= FAIM, 1-D MULTI-GP	DIFFUSION	SLAB CYL	SPHERE	CDC
RS	P T= FAIMOS, 1-D MULTI-GP	DIFFUSION	SLAB CYL	SPHERE	BOEC
T= EXTERMINATOR, 2-D MULTI-GP	DIFFUSION	PROGRAM	ORNL	7090 F+FAP	RS P
= EXTERMINATOR2, 2-D MULTI-GP	DIFFUSION	PROGRAM	ORNL	360 F4	RS P T
RI-QUIR, TIME-DEP 2-D MULTI-GP	DIFFUSION	WAL	7090 F+FAP	RS P T= VA	212
RS	P T= GASP2, 1-D FEW-GP	DIFFUSION	POWER	DIST SEARCH	GA
P	T= RELOAD-FEVER, 1-D FEW-GP	DIFFUSION	DEPLETION	GA	7044 F4
RSBP	T= GAMBLE4, 2-D MULTI-GP	DIFFUSION	XY RZ	GEOMETRY	GA
RS	P = GAMBLE5, 2-D MULTI-GP	DIFFUSION	XY RZ	GEOMETRY	GA
RS	P T= TEMCO, 1-D FEW-GP	DIFFUSION	TEMP	COEF CALC	GA
RS	P T= ASSAULT, 2-D MULTI-GP	DIFFUSION	DEPLETION	CODE	ORNL
RS	P T= HFN, 1-D MULTI-GP	DIFFUSION	SLAB CYL	SPHERE	BNWL
RS	P T= MACH1, 1-D MULTI-GP	DIFFUSION	SLAB CYL	SPHERE	ANL
P	T= VARI-QUIR3, 2-D MULTI-GP	DIFFUSION	XY RZ	RTH	WAL
CAESAR4/LIBLST, 1-D MULTI-GP	DIFFUSION	+ LIB	AI	360 F4	RSBP T=

RSBP T= THREDES, 1-D FEW-GP	DIFFUSION DESIGN SYSTEM	AI	360 F4	273
T= WIGL2, 1-D 2-GP SPACE-TIME	DIFFUSION 3-GEOM	BAPL 6600 F4	RS P	R274
T= PDQ7, 1,2 OR 3-D FEW-GP	DIFFUSION DEPLETION	BAPL 6600 F+ASC	RS	R275
6600 F4 RS T= M0807, 2-D	DIFFUSION ABSORPTION REMOVAL	X-SECS BAPL		R280
RS PLT= BISYN, 2-D MULTI-GP	DIFFUSION SYNTHESIS CALC	GESJ 635 F4		287
1-D MULTIGROUP TIME-DEPENDENT	DIFFUSION GGA	1108 F4 RS P T= GAKIN,		310
S P T= FEVER7, 1-D MULTIGROUP	DIFFUSION AND DEPLETION	GGA 1108 F4 R		318
RS P T= 2DB, 2-D MULTIGROUP	DIFFUSION AND DEPLETION	BNWL 1108 F4		325
L RS P T= PDQ5, 2-D FEW-GROUP	DIFFUSION AND DEPLETION	IBM 360 F+BA		R336
RS PLT= DTF, 1-D MULTIGROUP	DISCRETE ORDINATE CALC	UNC 1604 F63		144
RS PLT= DTF2, 1-D MULTI-GP	DISCRETE ORDINATE CALC	AI 7090 F2		151
RSBPLT= ANISN, 1-D MULTI-GP	DISCRETE ORDINATE CALC	AI 360 F4		151
3 RS = 2DF, 2-D MULTI-GP	DISCRETE ORDINATE CODE	UNC-LASL 1604 F6		173
RS P T= DTF4, 1-D MULTI-GP	DISCRETE ORDINATE PROGRAM	LASL 7030 F4		209
RS P T= DTF4, 1-D MULTI-GP	DISCRETE ORDINATE PROGRAM	LRC 7094 F4		209
RS P T= DTF4, 1-D MULTI-GP	DISCRETE ORDINATE PROGRAM	LASL 6600 F4		209
RS P T= DTF4, 1-D MULTI-GP	DISCRETE ORDINATE PROGRAM	BEC 625 F4		209
RS P T= DTF4, 1-D MULTI-GP	DISCRETE ORDINATE PROGRAM	ANL 360 F4		209
S P T= SNARG-1D, 1-D MULTI-GP	DISCRETE ORDINATE CALC	ANL 3600 F36 R		288
RS P T= TDSN, 2-D MULTIGROUP	DISCRETE ORDINATE PROGRAM	LRC 7090 F+MAP		312
= BURP2, DETECTOR EFFICIENCY	DISK SOURCE	UM 7090 MAD	RSB	165
GESJ 2000 F2 RS P =	DOPIE, RESOLVED RESONANCE	X-SECTION CALC		177
= SPARTA, SPATIALLY-AVERAGED	DOPPLER EFFECTS	GESJ 2000 F2 RS P		178
THUNDERHEAD, EXTERNAL+INTERNAL	DOSE AI	7094 F+MAP RS PLT=	CURIE/DOSE/	196
OE, SHIELD WEIGHT OPTIMIZATION	DOSE CALC	AI 7094 F2 RS P = SH		197
/DOSET, HTGR ACCIDENT ANALYSIS	DOSE CALC GA	1108 F4 RS T= GADOSE		261
2 RS P = CLOUD, GAMMA-RAY	DOSE RATE FROM A CLOUD	AI 7090 F		47
63 RS P = CLOUD, GAMMA-RAY	DOSE RATE FROM A CLOUD	CDC 1604 F		47
4 RS P = CLOUD, GAMMA-RAY	DOSE RATE FROM A CLOUD	SRL 360 F		47
7090 F2 RS = FARSE1A,	DOSE RATE FROM SNAP SHIELD LEAKAGE	AI		91
7090 F2 RS P = RATRAP,	DOSE RATE CALCULATION SNAP GEOMETRY	AI		141
7090 F2 RS = MORTIMER,	DOSE RATE CALCULATION SNAP GEOMETRY	AI		142
RS P = MIST, 1-D FEW-GP SN	DOUBLE SN APPROX SLAB GEOM	PP 7090 F2		59
T= GE-HAPO-S13, 1-D MULTI-GP	DOUBLE SN APPROX	BNWL 7090 FLOCO	RSBP	75
RAMS LASL 7090 F2 RS P T=	DPC, DATA PREPARATION FOR 2-D DESIGN	PROG		234
EFFECTIVE X-SECTION CALC FROM	DSN OUTPUT PW	1604 LAG1 RS P = DTX,		210
ALCULATION OF SN CONSTANTS FOR	DSN TDC PW	1604 F63 RS P = SNC, C		189
AI 7090 F2 RS PLT=	DTF2, 1-D MULTI-GP DISCRETE ORDINATE CALC			151
S P T= DTF-BURN, 1-D MULTI-GP	DTF4 WITH DEPLETION	LASL 7030 F4 R		269
RAM LASL 7030 F4 RS P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROG			209
RAM LRC 7094 F4 RS P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROG			209
RAM LASL 6600 F4 RS P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROG			209
RAM BEC 625 F4 RS P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROG			209
RAM ANL 360 F4 RS P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROG			209
N LASL 7030 F4 RS P T=	DTF-BURN, 1-D MULTI-GP DTF4 WITH DEPLETIO			269
C UNC 1604 F63 RS PLT=	DTF, 1-D MULTIGROUP DISCRETE ORDINATE CAL			144
TPUT PW 1604 LAG1 RS P =	DTX, EFFECTIVE X-SECTION CALC FROM DSN OU			210
APL 6600 F4 RS P T= M0552,	DYNAMIC ANALYSIS LINEAR ELASTIC SYSTEMS B			R283
AIROS2A, SIMULATION OF REACTOR	DYNAMICS	AI 360 F+BAL	RS P T=	326
NG BMI 6400 F4 RS P T=	ECCSA1, LOSS-OF-COOLANT + EMERGENCY COOLI			330
S = WED, W-DSN OUTPUT TAPE	EDIT REACTION RATES	UK-W 7090 F2 R		133
S PLT= AILMOE, X-SECTION CALC	ELASTIC SCAT RESONANCES	AI 7090 F+MAP R		147
S PLT= AILMOE, X-SECTION CALC	ELASTIC SCAT RESONANCES	AI 360 F4 R		147
M0552, DYNAMIC ANALYSIS LINEAR	ELASTIC SYSTEMS BAPL 6600 F4	RS P T=		R283
1108 F4 RS = HEXSCAT,	ELASTIC SCAT X-SECTIONS HEX LATTICE	GGA		291
00 F4 RS P T= M0457(PIPE),	ELASTIC STRESS OF PIPING SYSTEM	BAPL 66		R329
RNL 1604 F63 RS P = PEGGY,	ELASTIC SCATTERING PHASE-SHIFT ANALYSIS	D		334
GA 7044 F4 RS P T=	ELBOW, FUEL ELEMENT STRESS ANALYSIS	STUDY		229
T= ECCSA1, LOSS-OF-COOLANT +	EMERGENCY COOLING	BMI 6400 F4 RS P		330
SR, DATA FILE SERVICE ROUTINES	ENDF TAPES	BNL 7090 F+MAP	RS P T= DF	236
A 7044 F4 RS P = GAVER,	ENERGY INTERVAL X-SECTION AVERAGING	G		218

ATION GEV 7094 F+FAP RS P T=	EPITHERMOS, SPECTRUM AND X-SECTION CALCUL	201
P T= REAX, RESOLVED RESONANCE	EPITHERMAL X-SECTIONS GESJ 635 F4 RS	257
GA 1108 F4 S T= GAFFE,	EQUILIBRIUM FUEL CYCLE CALCULATION G	302
L ORNL 7090 F2 RS P T=	EQUIPOISE3, 2-D 2-GROUP DIFFUSION SLAB CY	39
SLAB ORNL 7090 F2 RS P T=	EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER	87
SLAB CDC 1604 F63 RS P T=	EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER	87
UPITER1, COUPLED-CHANNEL X-SEC	EVALUATION ORNL 1604 F63 RS P T= J	308
FLE, FISSILE NUCLIDE X-SECTION	EVALUATION ORNL 7090 F4 RSBP = MUF	323
RS P T= FCRE, FAST REACTOR	EXCURSION CALCULATIONS GESJ 2000 F4	174
RS P T= FORE2, FAST REACTOR	EXCURSION CALCULATIONS GESJ 635 F4	174
RS P T= FORE2, FAST REACTOR	EXCURSION CALCULATIONS ANL 360 F4	174
7094 F4 RS P = MARS, 2-D	EXCURSION CALCULATION R-Z GEOMETRY APDA	293
URVES LRL 7094 F2 RS P =	EXPALS, LEAST SQUARES EXPONENTIAL DECAY C	321
RS P T= CURFIT, CURVE FITTING	EXPERIMENTAL DATA POINTS KAPL 6600 F+ASC	R 43
P = CROC9C, ML-1 FLUID FLOW	EXPERIMENT ANALYSIS AGN 7090 F2 RS	154
2 RS PLT= HAMMER, CRITICAL	EXPERIMENT ANALYSIS SYSTEM BNL-SRL 7090 F	277
4 RS PLT= HAMMER, CRITICAL	EXPERIMENT ANALYSIS SYSTEM BNL-SRL 360 F	277
APDA 7094 F2 RS P = WEAK	EXPLOSION, COUPLED NEUTRON-HYDRODYNAMICS	145
ATA GESJ 635 F4 RS P T=	EXPN, ANALYSIS OF PULSED NEUTRON SOURCE D	258
RS P = EXPALS, LEAST SQUARES	EXPONENTIAL DECAY CURVES LRL 7094 F2	321
NTIC, LEAST SQUARES FIT SUM OF	EXPONENTIALS MIT 7090 F2 RS P = FRA	324
RAM ORNL 7090 F+FAP RS P T=	EXTERMINATOR, 2-D MULTI-GP DIFFUSION PROG	156
GRAM ORNL 360 F4 RS P T=	EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PRO	156
PLT= CURIE/DOSE/THUNDERHEAD,	EXTERNAL+INTERNAL DOSE AI 7094 F+FAP RS	196
TERS WAL 7094 F2 RS P =	EXT, X-SECTIONS FROM B-W RESONANCE PARAME	238
IMATION RZ GEOMETRY LRL 709 F2	RS = FIRN, 2-D FEW-GP S4 APPROX	7
SLAB CYLINDER SPHERE AI 7090 F2	RS P T= FOG, 1-D FEW-GP DIFFUSION	28
AIM AND FOG CODES AI 7090 F2	RS = PERT, 1-D PERTURBATION FOR	30
FUSION XYZ GEOMETRY ORNL 7090 F2	RS P T= WHIRLAWAY, 3-D 2-GROUP DIF	32
M X-SECTION CALC GA 7090 F2	RS PLT= GAM1, FAST NEUTRON SPECTRU	33
FFUSION SLAB CYL ORNL 7090 F2	RS P T= EQUIPOISE3, 2-D 2-GROUP DI	39
FUSION SLAB CYLINDER ORNL 7090 F2	RS P T= ZOGRAND, 2-D FEW-GROUP DIF	40
ION SLAB GEOMETRY AI 7090 F2	RS P = GRACE1, GAMMA-RAY ATTENUAT	45
ION CYL SPHERE GEOM AI 7090 F2	RS P = GRACE2, GAMMA-RAY ATTENUAT	46
FROM A CLOUD AI 7090 F2	RS P = CLOUD, GAMMA-RAY DOSE RATE	47
ATURE VOID FRACTION AI 7090 F2	RS = FUGUE, STEADY-STATE TEMPER	48
M X-SECTION CALC AI 7090 F2	RS PLT= FORM, FAST NEUTRON SPECTRU	51
ATION SLAB GEOMETRY AI 7090 F2	RS = SAIL, 1-D 1-GP SN APPROXIM	52
S4 APPROXIMATION AI 7090 F2	RS = S4 CYL CELL CODE, 1-D 1-GP	53
E COST ANALYSIS AI 7090 F2	RS P = AIMFIRE, URANIUM FUEL CYCL	55
RING KERNEL CALC GA 7090 F2	RS T= SUMMIT, CRYSTALLINE SCATTE	56
FUSION DEPLETION AI 7090 F2	RS LT= SIZZLE, 1-D MULTIGROUP DIF	58
SN APPROX SLAB GEOM PP 7090 F2	RS P = MIST, 1-D FEW-GP SN DOUBLE	59
USION CYLINDER SLAB ORNL 7090 F2	RS P T= EQUIPOISE3A, 2-D 2-GP DIFF	87
X-SECTION CALC AI 7090 F2	RS PLT= ARES2, RESONANCE INTEGRAL	89
AP SHIELD LEAKAGE AI 7090 F2	RS = FARSE1A, DOSE RATE FROM SN	91
TEMPEST COMBINATION AI 7090 F2	RS P = BAM, S4 CYL CELL CODE AND	108
TOR FINS SNAP GEOM AI 7090 F2	RS = SCARF2, SCATTER FROM RADIA	110
SNAP GEOMETRY AI 7090 F2	RS = SCAR1, SCATTER FROM A RING	111
ESIGN OPTIMIZATION AI 7090 F2	RS = CROCK, SPACE POWER PLANT D	112
ESIGN OPTIMIZATION AI 7090 F2	RS = SHOCK, SPACE POWER PLANT D	114
N DEPLETION PROGRAM GA 7090 F2	RSB T= FEVER, 1-D FEW-GP DIFFUSIO	117
KINETICS W/FEEDBACK AI 7090 F2	RS = AIREK3, SPACE-INDEPENDENT	121
TICS SNAP GEOMETRY AI 7090 F2	RS = SNAPKIN5/5A, 1-REGION KINE	122
TRATION CALCULATION DAC 7090 F2	RSBP = LIPRECAN1, MC NEUTRON PENE	123
HYDRODYNAMICS SPHERE LRL 7090 F2	RS P = CONEC, COUPLED NEUTRONICS-	129
ROX SLAB CYL SPHERE UK-W 7090 F2	RS P T= W-DSN, 1-D MULTI-GP SN APP	132
T REACTION RATES UK-W 7090 F2	RS = WED, W-DSN OUTPUT TAPE EDI	133
CHAIN EQUATIONS ORNL 7090 F2	RSBP = NUCY, SOLUTION OF NUCLIDE	134
S HETEROGENEOUS CALC FMA 7090 F2	RSBP = HERESY1, LATTICE PARAMETER	136
ION SNAP GEOMETRY AI 7090 F2	RS P = RATRAP, DOSE RATE CALCULAT	141

ATION SNAP GEOMETRY	AI	7090	F2	RS	=	MORTIMER, DOSE RATE CALCUL	142
OUP SLAB GEOMETRY	BNWL	7090	F2	RSBPLT=		MAC, SHIELD DESIGN MULTIGR	143
UTRON-HYDRODYNAMICS	APDA	7094	F2	RS P	=	WEAK EXPLOSION, COUPLED NE	145
PERFORMANCE DATA	KE	7090	F2	RSBP	=	NPRFCCP, FUEL CYCLE COSTS	146
XIMATION CYLINDER	PP	7090	F2	RS P T=		TOPIC, 1-D FEW-GP SN APPRO	148
E CHCRD DIST FUNCT	AEG	7090	F2	RS	=	DANCOFF JR, MODERATOR SPAC	150
E ORDINATE CALC	AI	7090	F2	RS PLT=		DTF2, 1-D MULTI-GP DISCRET	151
S-HYDRODYNAMICS CODE	AGN	7090	F2	RSB	=	HATCHET, COUPLED NEUTRONIC	153
PERIMENT ANALYSIS	AGN	7090	F2	RS P	=	CROC90, ML-1 FLUID FLOW EX	154
MPERATURE HISTORY	KE	7094	F2	RSBP T=		PTH1, BLOWDOWN PRESSURE TE	155
L X-SECTION CALC	GESJ	2000	F2	RS P	=	RAPTURE, RESONANCE INTEGRA	176
X-SECTION CALC	GESJ	2000	F2	RS P	=	DOPIE, RESOLVED RESONANCE	177
DOPPLER EFFECTS	GESJ	2000	F2	RS P	=	SPARTA, SPATIALLY-AVERAGED	178
CHAIN ANALYSIS	ORNL	7090	F2	RS P	=	ISOCRUNCH, REACTION DECAY	180
N FUEL TUBE BUNDLES	LASL	7094	F2	RS P	=	AXFLU, HEAT TRANSFER MOLTE	182
D FUEL TUBE BUNDLES	LASL	7094	F2	RS P	=	AXTHRM, HEAT TRANSFER SOLI	183
ZATION DOSE CALC	AI	7094	F2	RS P	=	SHOE, SHIELD WEIGHT OPTIMI	197
-STATE HEAT TRANSFER	AI	7094	F2	RS P T=		HEATING2, TRANSIENT STEADY	198
TI-GP CONSTANT CALC	AGN	7090	F2	RSBPLT=		AGN-GAM, FAST SPECTRUM MUL	204
2-D DESIGN PROGRAMS	LASL	7090	F2	RS P T=		DPC, DATA PREPARATION FOR	234
-PIN FUEL ELEMENT	AGN	7090	F2	RS P	=	BOUNCE, FLUX DIST IN MULTI	237
ESONANCE PARAMETERS	WAL	7094	F2	RS P	=	EXT, X-SECTIONS FROM B-W R	238
SRS X-SECTION TAPES	WAL	7094	F2	RS T=		CPS, SC4020 PLOTS FROM SCI	239
P TRANSFER MATRICES	AGN	7090	F2	RS P T=		AGN-SIGMA, CALC OF MULTI-G	243
CLE ANALYSIS	AI	7094	F2	RS PLT=		CYCLOPS1, THERMODYNAMIC CY	244
2-D 2-PHASE FLOW	AI	7094	F2	RS P	=	FLOW-MODEL, MULTI-CHANNEL	246
AL MDEL X-SECTIONS	AI	7094	F2	RS P T=		2PLUS, NON-SPHERICAL OPTIC	254
OW AND HEAT TRANSFER	AI	7094	F2	RS P T=		TRANS-FUGUE1, TRANSIENT FL	268
T ANALYSIS SYSTEM BNL-SRL	7090	F2	RS PLT=		HAMMER, CRITICAL EXPERIMEN	277	
IRING LISTS UHTREX	LASL	7090	F2	RS P	=	WIREX, COMPUTER-PRODUCED W	315
MENTIAL DECAY CURVES	LRL	7094	F2	RS P	=	EXPALS, LEAST SQUARES EXPO	321
SUM OF EXPONENTIALS	MIT	7090	F2	RS P	=	FRANTIC, LEAST SQUARES FIT	324
E ABSORPTION, CELL	ANL	3600	F36	RSBP T=		RABBLE/WLIB/FLAT, RESONANC	281
CRETE ORDINATE CALC	ANL	3600	F36	RS P T=		SNARG-1D, 1-D MULTI-GP DIS	288
LAB CYLINDER SPHERE	ROA	360	F4	RS PLT=		FIRE2, 1-D AGE-DIFFUSION S	9
AIM AND FOG CODES	BOEC	360	F4	RS P	=	PERT, 1-D PERTURBATION FOR	30
FROM A CLOUD	SRL	360	F4	RS P	=	CLOUD, GAMMA-RAY DOSE RATE	47
SPECTRUM X-SECTIONS	BOEC	360	F4	RS PLT=		TEMPEST2, THERMAL NEUTRON	50
FUSION DEPLETION	AI	360	F4	RSBP T=		SIZZLE, 1-D MULTIGROUP DIF	58
L CURVE FITTING	LASL	7094	F4	RS	=	LASL LEAST SQUARES, GENERA	62
LS OF REVOLUTION	KAPL	6600	F4	RS P T=		SOR2, STRESS ANALYSIS SHEL	R 80
SION SLAB CYL SPHERE	BOEC	360	F4	RS P T=		FAIMOS, 1-D MULTI-GP DIFFU	120
KINETICS W/FEEDBACK	ROA	360	F4	RS P	=	AIREK3, SPACE-INDEPENDENT	121
STIC SCAT RESONANCES	AI	360	F4	RS PLT=		AILMOE, X-SECTION CALC ELA	147
XIMATION CYLINDER	PP	7040	F4	RS P T=		TOPIC, 1-D FEW-GP SN APPRO	148
TE ORDINATE CALC	AI	360	F4	RSBPLT=		ANISN, 1-D MULTI-GP DISCRE	151
P DIFFUSION PROGRAM	ORNL	360	F4	RS P T=		EXTERMINATOR2, 2-D MULTI-G	156
POWER DISTRIBUTION	GESJ	635	F4	RS P T=		FLARE, 3-D REACTIVITY AND	167
VERSE KINETICS CALC	WAL	7094	F4	RS	=	R102, SPACE-INDEPENDENT IN	168
EQUIREMENT STUDY	AI	360	F4	RS P T=		AISITE2, PARAMETRIC SITE R	172
ON CALCULATIONS	GESJ	2000	F4	RS P T=		FORE, FAST REACTOR EXCURS	174
ION CALCULATIONS	GESJ	635	F4	RS P T=		FORE2, FAST REACTOR EXCURS	174
ION CALCULATIONS	ANL	360	F4	RS P T=		FORE2, FAST REACTOR EXCURS	174
CALC 0 TO 10 MEV	BNWL	1107	F4	RS PLT=		GAMTEC2, MULTI-GP CONSTANT	185
-LEVEL CONVOLUTION	PP	7040	F4	RS P	=	COMBCO, COMBINED B-W MULTI	203
TH 2-D HEAT TRANSFER	GA	7044	F4	RS T=		BLOOST5, POINT-KINETICS WI	205
OM X-SECTION TAPES	GA	7044	F4	RSBP T=		CROSSPLOT, SC4020 PLOTS FR	207
E ORDINATE PROGRAM	LASL	7030	F4	RS P T=		DTF4, 1-D MULTI-GP DISCRET	209
E ORDINATE PROGRAM	LRC	7094	F4	RS P T=		DTF4, 1-D MULTI-GP DISCRET	209
E ORDINATE PROGRAM	LASL	6600	F4	RS P T=		DTF4, 1-D MULTI-GP DISCRET	209
E ORDINATE PRCGRAM	BEC	625	F4	RS P T=		DTF4, 1-D MULTI-GP DISCRET	209

E ORDINATE PROGRAM	ANL	360	F4	RS P T=	DTF4, 1-D MULTI-GP DISCRET	209
W RESONANCE DATA	GA	7044	F4	RS P =	FASDOP, X-SECTIONS FROM B-	216
LC FOR ANGULAR DIST	GA	7044	F4	RS P =	LEGCOEF3, LEGENDRE COEF CA	217
ECTION AVERAGING	GA	7044	F4	RS P =	GAVER, ENERGY INTERVAL X-S	218
N POWER DIST SEARCH	GA	7044	F4	RS P T=	GASP2, 1-D FEW-GP DIFFUSIO	220
IFFUSION DEPLETION	GA	7044	F4	RS P T=	RELOAD-FEVER, 1-D FEW-GP D	221
ALYSIS W/REFUELING	GA	7044	F4	RS P T=	REVISED-GAD, FUEL CYCLE AN	223
PERFORMANCE STUDY	GA	7044	F4	RS P =	WAMPUM, FUEL CYCLE COSTS P	224
N TEMP COEF CALC	GA	7044	F4	RS P T=	TEMCO, 1-D FEW-GP DIFFUSIO	225
SIGN CRITERIA COSTS	GA	7044	F4	RS P T=	STMGEN, STEAM GENERATOR DE	227
RESSURE POST RUPTURE	GA	7044	F4	RS T=	PRECON, HTGR CONTAINMENT P	228
ANALYSIS STUDY	GA	7044	F4	RS P T=	ELBOW, FUEL ELEMENT STRESS	229
IS OF 2-D BODIES	GA	7044	F4	RS P T=	PIZZA, PLANE STRESS ANALYS	230
ACTIVITY DIST STUDY	GA	7044	F4	RS P =	RAD2, HTGR FISSION PRODUCT	231
THERMAL DESIGN STUDY	GA	7044	F4	RS P =	GAZELLE5, GAS-COOLED CORE	232
UEL TEMPERATURE CODE	GA	7044	F4	RS =	CORE, CORE CONFIGURATION F	233
IGNS FOR GAMMA-RAYS	GEC	7094	F4	RS P =	GAMMA-P, PRODUCTION X-SECT	235
N SLAB CYL SPHERE	BNWL	1107	F4	RS P T=	HFN, 1-D MULTI-GP DIFFUSIO	241
SECTION CALCULATION	GA	7044	F4	RS =	FLANGE1, SCATTERING LAW X-	247
AXISYMMETRIC LOAD	GA	7044	F4	RS P T=	SAFE-PCRS, STRESS ANALYSIS	250
IS AXISYMMETRIC LOAD	GA	7044	F4	RS P T=	SAFE-AXISYM, STRESS ANALYS	251
NALYSIS, 2-D BODIES	GA	1108	F4	RS P =	SAFE-PLANE, PLANE STRESS A	252
S THIN SHELLS	GA	1108	F4	RS P =	SAFE-SHELL, STRESS ANALYSI	253
NETICS KEX OPTIONS	WAL	7094	F4	RS P =	R101, SPACE-INDEPENDENT KI	255
L-HYDRAULIC ANALYSIS	GESJ	635	F4	RS P T=	MANTA, STEADY-STATE THERMA	256
PITHERMAL X-SECTIONS	GESJ	635	F4	RS P T=	REAX, RESOLVED RESONANCE E	257
EUTRON SOURCE DATA	GESJ	635	F4	RS P T=	EXPAN, ANALYSIS OF PULSED N	258
ING CODE CAVITY GEOM	GA	1108	F4	RS P T=	MUSCAT, VIEW FACTOR SHIELD	259
SIS PARTIAL REFUEL	GA	7044	F4	RS P T=	GARGOYLE, FUEL CYCLE ANALY	260
T ANALYSIS DOSE CALC	GA	1108	F4	RS T=	GADOSE/DOSET, HTGR ACCIDEN	261
LAW CALCULATION	GA	1108	F4	RS P =	GASKET, THERMAL SCATTERING	263
IFFUSION XY RZ RTH	WAL	7094	F4	RS P T=	VARI-QUIR3, 2-D MULTI-GP D	264
L FUEL ELEMENT	BAPL	6600	F4	RS T=	CYGRO2, STRESS ANALYSIS CY	R266
4,500PSIA 32-472DEGF	BAPL	2000	F4	RS =	WATER, STEAM TABLES 14.5-1	R267
4 WITH DEPLETION	LASL	7030	F4	RS P T=	DTF-BURN, 1-D MULTI-GP DTF	269
GP DIFFUSION + LIB	AI	360	F4	RSBPLT=	CAESAR4/LIBLST, 1-D MULTI-	270
ARY UTILITY ROUTINE	AI	360	F4	RSB LT=	CLIP, FORM OR THREDES LIBR	271
TEMPERATURE STUDY	BAPL	6600	F4	RS P T=	FIGRO, LSBR FUEL SWELLING	R272
ION DESIGN SYSTEM	AI	360	F4	RSBP T=	THREDES, 1-D FEW-GP DIFFUS	273
DIFFUSION 3-GEOM	BAPL	6600	F4	RS P T=	WIGL2, 1-D 2-GP SPACE-TIME	R274
ION CALCULATION	GEC	625	F4	RS P =	AVOID, ANNULAR VOID X-SECT	276
T ANALYSIS SYSTEM BNL	SRL	360	F4	RS PLT=	HAMMER, CRITICAL EXPERIMEN	277
SYSTEM ANALYSIS	KE	3600	F4	RSBP =	WHAM, LIQUID-FILLED PIPING	278
OWN ANALYSIS LOFT	UG	360	F4	RS P =	WATER-HAMMER, LIQUID BLOWD	278
PTION REMOVAL X-SECS	BAPL	6600	F4	RS T=	M0807, 2-D DIFFUSION ABSOR	R280
NALYSIS AXISYM LOAD	BAPL	6600	F4	RS P T=	SEALSHL2, SHELL STRESS A	R282
NEAR ELASTIC SYSTEMS	BAPL	6600	F4	RS P T=	M0552, DYNAMIC ANALYSIS LI	R283
NT ACCIDENT ANALYSIS	BAPL	6600	F4	RS P T=	M0555(ACT1), LOSS-OF-COOLA	R284
E INTEGRAL HEX CELL	BAPL	6600	F4	RS P T=	RESQ2/RESQ0/DBF1, RESONANC	R285
ONDUCTION PROGRAM	BAPL	6600	F4	RS P T=	HOT2, 2-D TRANSIENT HEAT C	R286
ION SYNTHESIS CALC	GESJ	635	F4	RS PLT=	BISYN, 2-D MULTI-GP DIFFUS	287
CTION CALC MODERATOR	GGA	1108	F4	RS P =	GAKER, INELASTIC SCAT X-SE	289
EACTOR KINETICS EQNS	GGA	1108	F4	RS P =	GASA, STABILITY ANALYSIS R	290
CTIONS HEX LATTICE	GGA	1108	F4	RS =	HEXSCAT, ELASTIC SCAT X-SE	291
NCE PARAMETER CALC	GGA	1108	F4	RS P =	PSEUDO, STATISTICAL RESONA	292
ATION R-Z GEOMETRY	APDA	7094	F4	RS P =	MARS, 2-D EXCURSION CALCUL	293
.5-2538 PSIA	BAPL	6600	F4	RS P T=	M0899/HOH, STEAM TABLES 14	R294
CIDENT ANALYSIS	BAPL	6600	F4	RS P T=	FLASH2, LOSS-OF-COOLANT AC	R295
R SAFE PROGRAMS	GGA	1108	F4	RS P =	GRDWRK, GRID GENERATION FO	296
FAST THERMAL SPECTRA	GGA	1108	F4	RS PLT=	GGC4, MULTI-GP X-SECTIONS	298
RIBUTION PROGRAM	KAPL	6600	F4	RS P T=	LION, 3-D TEMPERATURE DIST	R299

NALYSIS CONCRETE	GGA	1108	F4	RS P =	SAFE-CREEP, VISCOELASTIC A	300
SION PRODUCT RELEASE	GGA	1108	F4	RS P =	FREVAP6, HTGR METALLIC FIS	301
CLE CALCULATION	GGA	1108	F4	S T=	GAFFE, EQUILIBRIUM FUEL CY	302
2-D HEAT TRANSFER	GGA	1108	F4	RS P T=	BLOOST6, COMBINED KINETICS	303
RZ RTHETA GEOMETRY	BNWL	1108	F4	RS P =	PERT4, 2-D PERTURBATION XY	304
INTEGRAL CALCULATION	BW	2000	F4	RS =	STRIP, RESOLVED RESONANCE	305
T REACTOR X-SEC CALC	BNWL	1108	F4	RS P T=	FCC4, FUNDAMENTAL MODE FAS	306
O CELL CALCULATION	AI	360	F4	RS P T=	HWOCR-SAFE, 2-D MONTE CARL	307
EACTOR KINETICS	AI	7094	F4	RS P T=	TSN, SPATIALLY-DEPENDENT R	309
EACTOR KINETICS	AI	360	F4	RS P T=	TSN, SPATIALLY-DEPENDENT R	309
-DEPENDENT DIFFUSION	GGA	1108	F4	RS P T=	GAKIN, 1-D MULTIGROUP TIME	310
OPIC BURNUP ANALYSIS	GEV	635	F4	RS P T=	BURNUP, HEAVY ELEMENT ISOT	311
TION FISSION PRODUCT	BAPL	6600	F4	RS P =	CINDER(MO102), POINT DEPLE	313
-RAY RADIOACTIVITY	IITR	7094	F4	RSBPLT=	NAP, NEUTRON-INDUCED GAMMA	314
-AVERAGED X-SEC CALC	GGA	1108	F4	RS P T=	GAF/GAR, SPECTRA AND GROUP	316
T REACTOR KINETICS	GGA	1108	F4	RS P =	GAPOTKIN, SPACE-INDEPENDEN	317
FUSION AND DEPLETION	GGA	1108	F4	RS P T=	FEVER7, 1-D MULTIGROUP DIF	318
STRIBUTION SEARCH	GGA	1108	F4	RS P T=	GASP7, 1-D BURNUP POWER DI	319
CIENT CALCULATION	GGA	1108	F4	RS P T=	TEMCO7, TEMPERATURE COEFFI	320
ION FLUX, X-SEC CALC	ORNL	360	F4	RS P =	ISOSEARCH, ISOTOPE PRODUCT	322
SECTION EVALUATION	ORNL	7090	F4	RSBP =	MUFFLE, FISSION NUCLEIDE X-	323
ION AND DEPLETION	BNWL	1108	F4	RS P T=	2DB, 2-D MULTIGROUP DIFFUS	325
ISSILE NUCLEIDE DATA	BAPL	6600	F4	RS P T=	DAFT1, LEAST SQUARES FIT F	R327
T THERMAL ANALYSIS	BMI	6400	F4	RS P T=	NURLOC-1.0, LOSS-OF-COOLAN	328
S OF PIPING SYSTEM	BAPL	6600	F4	RS P T=	MO457(PPIPE), ELASTIC STRES	R329
EMERGENCY COOLING	BMI	6400	F4	RS P T=	ECCSA1, LOSS-OF-COOLANT +	330
NSIENT ANALYSIS	BAPL	6600	F4	RS P T=	FLOT1(MO219), PWR FLOW TRA	R331
UCTURE STRESS STUDY	GGA	1108	F4	RS P T=	SAFE-3D, 3-D COMPOSITE STR	332
ER GAMMA-RAY SPECTRA	GGA	1108	F4	RS P T=	TOAD, PROCESSING OF ANALYZ	333
NELASTIC SCATTERING	CDC	1604	F63	RS P =	HAFEVER, HAUSER-FESHBACH I	14
SLAB CYLINDER SPHERE	CDC	1604	F63	RS P T=	FOG, 1-D FEW-GP DIFFUSION	28
ON SLAB CYL SPHERE	CDC	1604	F63	RS PLT=	AIM6, 1-D MULTI-GP DIFFUSI	29
AIM AND FOG CODES	CDC	1604	F63	RS =	PERT, 1-D PERTURBATION FOR	30
FUSION XYZ GEOMETRY	CDC	1604	F63	RS P T=	WHIRLAWAY, 3-D 2-GROUP DIF	32
M X-SECTION CALC	CDC	1604	F63	RS PLT=	GAM1, FAST NEUTRON SPECTRU	33
M X-SECTION CALC	ANL	3600	F63	RS LT=	GAM1, FAST NEUTRON SPECTRU	33
FUSION SLAB CYLINDER	CDC	1604	F63	RS P T=	20GRAND, 2-D FEW-GROUP DIF	40
ION SLAB GEOMETRY	CDC	1604	F63	RS P =	GRACE1, GAMMA-RAY ATTENUAT	45
ION CYL SPHERE GEOM	CDC	1604	F63	RS P =	GRACE2, GAMMA-RAY ATTENUAT	46
FROM A CLOUD	CDC	1604	F63	RS P =	CLOUD, GAMMA-RAY DOSE RATE	47
SPECTRUM X-SECTIONS	CDC	1604	F63	RS PLT=	TEMPEST2, THERMAL NEUTRON	50
M X-SECTION CALC	CDC	1604	F63	RS PLT=	FORM, FAST NEUTRON SPECTRU	51
S4 APPROXIMATION	CDC	1604	F63	RS =	S4 CYL CELL CODE, 1-D 1-GP	53
FUSION DEPLETION	CDC	1604	F63	RS LT=	SIZZLE, 1-D MULTIGROUP DIF	58
USION CYLINDER SLAB	CDC	1604	F63	RS P T=	EQUIPOISE3A, 2-D 2-GP DIF	87
X-SECTION CALC	CDC	1604	F63	RS PLT=	ARES2, RESONANCE INTEGRAL	89
DRODYNAMICS SPHERE	CDC	3600	F63	RSBP =	AX1, COUPLED NEUTRONICS-HY	102
ON SLAB CYL SPHERE	CDC	1604	F63	RS PLT=	FAIM, 1-D MULTI-GP DIFFUSI	120
CHAIN EQUATIONS	ORNL	1604	F63	RSBP =	NUCY, SOLUTION OF NUCLEIDE	134
FUNCTION SYNTHESIS	ANL	3600	F63	RSBP T=	TRAFICORPORATION, TRANSFER	135
ROUP SLAB GEOMETRY	ANL	3600	F63	RSBPLT=	MAC, SHIELD DESIGN MULTIGR	143
TE ORDINATE CALC	UNC	1604	F63	RS PLT=	DTF, 1-D MULTIGROUP DISCRE	144
RE CALC CYLINDER	ANL	3600	F63	RSBP T=	ARGUS, TRANSIENT TEMPERATU	152
SN APPROXIMATION RZ	PW	1604	F63	RS P T=	FORTTRAN TDC, 2-D MULTI-GP	161
POWER DISTRIBUTION	CDC	3600	F63	RSBP T=	FLARE, 3-D REACTIVITY AND	167
VERSE KINETICS CALC	ANL	3600	F63	RSBP =	R102, SPACE-INDEPENDENT IN	168
X-SECTION CALC	ANL	3600	F63	RSBP =	NEARREX, COMPOUND NUCLEUS	171
IREMENT STUDY	ANL	3600	F63	RS P T=	R153, PARAMETRIC SITE REQU	172
ORDINATE CODE	UNC-LASL	1604	F63	RS =	2DF, 2-D MULTI-GP DISCRETE	173
OM REACTION OR DECAY	ORNL	1604	F63	RS P =	ISOTOPES, MAXIMUM YIELD FR	179
CHAIN ANALYSIS	ORNL	1604	F63	RS P =	ISOCRUNCH, REACTION DECAY	180

UTILITY ROUTINE	ANL	3600	F63	RSBP T=	XLIBIT, X-SECTION LIBRARY	181
TRUM X-SECTION CALC	ANL	3600	F63	RS P T=	THERMOS(ANL), THERMAL SPEC	184
LLER DESIGN STUDY	PW	1604	F63	RS P T=	PIP, CENTRIFUGAL PUMP IMPE	187
EVALUATION	PW	1604	F63	RS P =	CMPXMAT, TRANSFER FUNCTION	188
STANTS FOR DSN TDC	PW	1604	F63	RS P =	SNC, CALCULATION OF SN CON	189
PTIMIZATION STUDY	PW	1604	F63	RS =	WOPEXPRT, REACTOR WEIGHT O	190
-HYDRODYNAMICS SPH	PW	1604	F63	RS P =	AX-TNT, COUPLED NEUTRONICS	191
YNTHESIS PROGRAM CYL	PW	1604	F63	RS T=	2DXYL, 3-D MULTI-GP FLUX S	192
Y TAPE PREPARATION	PW	1604	F63	RS =	CSP2A, SN X-SECTION LIBRAR	193
TAPE PREPARATION	PW	1604	F63	RS =	CSP1, SN X-SECTION LIBRARY	194
OR 2CXY FLUX INPUT	PW	1604	F63	RS =	TDP, 2-D PERTURBATION TDC	199
RAL CALC 2-REG CELL	ANL	3600	F63	RSBP T=	RIFF-RAFF, RESONANCE INTEG	213
RAL CALC HOMOGNEOUS	ANL	3600	F63	RSBP =	MISH-MASH, RESONANCE INTEG	214
G DATA REDUCTION	ANL	3600	F63	RSBP =	COINC, COINCIDENCE COUNTIN	248
AL MDEL X-SECTIONS	ANL	3600	F63	RSBP T=	2PLUS, NON-SPHERICAL OPTIC	254
NETICS KEX OPTIONS	ANL	3600	F63	RSBP =	R101, SPACE-INDEPENDENT KI	255
ICN SLAB CYL SPHERE	ANL	3600	F63	RS P T=	MACH1, 1-D MULTI-GP DIFFUS	262
X-SEC EVALUATION	ORNL	1604	F63	RS P T=	JUPITOR1, COUPLED-CHANNEL	308
ION FLUX, X-SEC CALC	ORNL	1604	F63	RS P =	ISOSEARCH, ISOTOPE PRODUCT	322
PHASE-SHIFT ANALYSIS	ORNL	1604	F63	RS P =	PEGGY, ELASTIC SCATTERING	334
ICN RADIAL INTEGRALS	ORNL	1604	F63	RS P =	RAMES, PARTICLE WAVE FUNCT	335
RIMENTAL DATA POINTS	KAPL	6600	F+ASC	RS P T=	CURFIT, CURVE FITTING EXPE	R 43
FFUSION DEPLETION	BAPL	6600	F+ASC	RS T=	PDQ7, 1,2 OR 3-D FEW-GP DI	R275
USION XY RZ GEOMETRY	GA	1108	F+BAL	RS P =	GAMBLE5, 2-D MULTI-GP DIFF	222
CTOR DYNAMICS	AI	360	F+BAL	RS P T=	AIROS2A, SIMULATION OF REA	326
ION AND DEPLETION	IBM	360	F+BAL	RS P T=	PDQ5, 2-D FEW-GROUP DIFFUS	R336
INSTRUCTION SET	PW	1604	F+CDP	RS P T=	LAG, ASSEMBLER FOR FLOCO2	186
AMMA-RAY HISTORIES	PW	1604	F+CDP	RS P T=	FMC-N/FMC-G, MC NEUTRON, G	195
ON SLAB CYL SPHERE	AI	7090	F+FAP	RS PLT=	AIM6, 1-D MULTI-GP DIFFUSI	29
ANCE INTEGRAL CALC	GA	7090	F+FAP	RSB T=	ZUT, RESOLVED REGION RESON	41
ONANCE INTEGRAL CALC	GA	7090	F+FAP	RSB T=	TUZ, UNRESOLVED REGION RES	42
SPECTRUM X-SECTIONS	AI	7090	F+FAP	RS PLT=	TEMPEST2, THERMAL NEUTRON	50
BURNUP RZ GEOMETRY	GA	7090	F+FAP	RSBP T=	DOB, 2-D FEW-GP DIFFUSION	99
DIFFUSION PROGRAM	UK-R	7090	F+FAP	RSBPLT=	CRAM, 1-D AND 2-D MULTI-GP	103
XIMUM MOMENT CALC	AI	7090	F+FAP	RS T=	4RESTRAINT PIPE STRESS, MA	109
ICN SLAB CYL SPHERE	AI	7090	F+FAP	RS LT=	ULCER, 1-D MULTI-GP DIFFUS	118
PECTRUM X-SECTIONS	AI	7090	F+FAP	RS LT=	QUICKIE, INFINITE MEDIUM S	119
ON SLAB CYL SPHERE	AI	7090	F+FAP	RS LT=	FAIM, 1-D MULTI-GP DIFFUSI	120
STIC SCAT RESONANCES	AI	7090	F+FAP	RS PLT=	AILMOE, X-SECTION CALC ELA	147
G-DOWN DENSITY CALC	AI	7094	F+FAP	RS PLT=	TYCHE3, MONTE CARLO SLOWIN	149
DIFFUSION PROGRAM	ORNL	7090	F+FAP	RS P T=	EXTERMINATOR, 2-D MULTI-GP	156
TILITY PROGRAM	AI	7094	F+FAP	RS LT=	MOMUS, X-SECTION LIBRARY U	159
NSFER COEFFICIENTS	LRL	7090	F+FAP	RS P T=	SOPHIST1/2/5, MULTI-GP TRA	160
ECTIONS PARAMETERS	AI	7094	F+FAP	RS PLT=	GRAVE, GROUP-AVERAGING X-S	162
EQUIREMENT STUDY	AI	7094	F+FAP	RS P T=	AISITE2, PARAMETRIC SITE R	172
CALC O TO 10 MEV	BNWL	7090	F+FAP	RS PLT=	GAMTEC2, MULTI-GP CONSTANT	185
TERNAL+INTERNAL DOSE	AI	7094	F+FAP	RS PLT=	CURIE/DOSE/THUNDERHEAD, EX	196
-SECTION CALCULATION	GEV	7094	F+FAP	RS P T=	EPITHERMOS, SPECTRUM AND X	201
LTI-GP DIFFUSION	WAL	7090	F+FAP	RS P T=	VARI-QUIR, TIME-DEP 2-D MU	212
UTINES ENDF TAPES	BNL	7090	F+FAP	RS P T=	DFSR, DATA FILE SERVICE RO	236
USION DEPLETION CODE	ORNL	7090	F+FAP	RS P T=	ASSAULT, 2-D MULTI-GP DIFF	240
ON WITH DEPLETION	WAPD	7090	F+FAP	RSBPLT=	LEOPARD, SPECTRA CALCULATI	279
INETICS W/FEEDBACK	AI	7094	F+MAP	RS P T=	AIROS, SPACE-INDEPENDENT K	163
OWN ANALYSIS LOFT	KE	7094	F+MAP	RSBP T=	SATURATED BLOWDOWN1, BLOWD	200
SRS X-SECTION TAPE	GA	7044	F+MAP	RSBP =	UNPACK, RETRIEVAL FROM SCI	206
X-SECTION CALC	AI	7094	F+MAP	RS PLT=	TRIX1, RESONANCE INTEGRAL	208
OR ANGULAR DIST DATA	AI	7094	F+MAP	RS P =	CHAD, LEGENDRE COEF CALC F	215
ND LATTICE EFFECTS	GA	7044	F+MAP	RS P T=	GAROL, RESONANCE OVERLAP A	219
USION XY RZ GEOMETRY	GA	7044	F+MAP	RSBP T=	GAMBLE4, 2-D MULTI-GP DIFF	222
NCE AND PRICE STUDY	GA	7044	F+MAP	RS P T=	OPUS, POWER PLANT PERFORMA	226
BURNUP CYL LATTICE	WAPD	7094	F+MAP	RS PLT=	LASER, SPECTRUM CALC WITH	249

ANALYSIS PROGRAM	PP	7040	F+MAP RS P T=	RSAC, RADIOLOGICAL SAFETY	265
ACCIDENT ANALYSIS	PPC	7040	F+MAP RS P T=	CONTEMPT, LOSS-OF-COOLANT	297
ETE ORDINATE PROGRAM	LRC	7090	F+MAP RS P T=	TDSN, 2-D MULTIGROUP DISCR	312
PHERE BOEC	360 F4	RS P T=	FAIMOS, 1-D MULTI-GP DIFFUSION SLAB CYL S		120
ERE AI	7090 F+FAP RS	LT=	FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPH		120
ERE CDC	1604 F63	RS PLT=	FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPH		120
CTION LUMPED MASS	LASL 7094	FAP RS P T=	RATH, 2- OR 3-D HEAT CONDU		242
GE AI	7090 F2	RS =	FARSE1A, DOSE RATE FROM SNAP SHIELD LEAKA		91
A GA	7044 F4	RS P =	FASDOP, X-SECTIONS FROM B-W RESONANCE DAT		216
GA 7090 F2	RS PLT=	GAM1,	FAST NEUTRON SPECTRUM X-SECTION CALC		33
CDC 1604 F63	RS PLT=	GAM1,	FAST NEUTRON SPECTRUM X-SECTION CALC		33
ANL 3600 F63	RS LT=	GAM1,	FAST NEUTRON SPECTRUM X-SECTION CALC		33
AI 7090 F2	RS PLT=	FORM,	FAST NEUTRON SPECTRUM X-SECTION CALC		51
CDC 1604 F63	RS PLT=	FORM,	FAST NEUTRON SPECTRUM X-SECTION CALC		51
GESJ 2000 F4	RS P T=	FORE,	FAST REACTOR EXCURSION CALCULATIONS		174
ESJ 635 F4	RS P T=	FORE2,	FAST REACTOR EXCURSION CALCULATIONS	G	174
NL 360 F4	RS P T=	FORE2,	FAST REACTOR EXCURSION CALCULATIONS	A	174
S P T= FCC4, FUNDAMENTAL MODE			FAST REACTOR X-SEC CALC BNWL 1108 F4	R	306
7090 F2	RSBPLT=	AGN-GAM,	FAST SPECTRUM MULTI-GP CONSTANT CALC	AGN	204
LT= GGC4, MULTI-GP X-SECTIONS			FAST THERMAL SPECTRA GGA 1108 F4	RS P	298
CALC BNWL 1108 F4	RS P T=	FCC4, FUNDAMENTAL MODE	FAST REACTOR X-SEC		306
ETION GGA 1108 F4	RS P T=	FEVER7, 1-D MULTIGROUP DIFFUSION AND DEPL			318
GRAM GA 7090 F2	RSB T=	FEVER, 1-D FEW-GP DIFFUSION DEPLETION PRO			117
709 F2	RS =	FIRN, 2-D FEW-GP S4 APPROXIMATION RZ GEOMETRY LRLL			7
7090 F2	RS P T=	FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE AI			28
1604 F63	RS P T=	FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE CDC			28
7090 F2	RS P =	MIST, 1-D FEW-GP SN DOUBLE SN APPROX SLAB GEOM PP			59
7090 F+FAP	RSBP T=	DDB, 2-D FEW-GP DIFFUSION BURNUP RZ GEOMETRY GA			99
7090 F2	RSB T=	FEVER, 1-D FEW-GP DIFFUSION DEPLETION PROGRAM	GA		117
7040 F4	RS P T=	TOPIC, 1-D FEW-GP SN APPROXIMATION CYLINDER	PP		148
7090 F2	RS P T=	TOPIC, 1-D FEW-GP SN APPROXIMATION CYLINDER	PP		148
7044 F4	RS P T=	GASP2, 1-D FEW-GP DIFFUSION POWER DIST SEARCH	GA		220
RS P T= RELOAD-FEVER, 1-D		FEW-GP DIFFUSION DEPLETION	GA 7044 F4		221
7044 F4	RS P T=	TEMCO, 1-D FEW-GP DIFFUSION TEMP COEF CALC	GA		225
60 F4	RSBP T=	THREDES, 1-D FEW-GP DIFFUSION DESIGN SYSTEM	AI 3		273
+ASC RS T=	PDQ7, 1,2 OR 3-D	FEW-GP DIFFUSION DEPLETION	BAPL 6600 F	R	275
90 F2	RS P T=	20GRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER ORNL 70			40
04 F63	RS P T=	20GRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER CDC 16			40
360 F+BAL	RS P T=	PDQ5, 2-D FEW-GROUP DIFFUSION AND DEPLETION	IBM		R336
DY BAPL 6600 F4	RS P T=	FIGRO, LSBR FUEL SWELLING TEMPERATURE STU			R272
7090 F+FAP	RS P T=	DFSR, DATA	FILE SERVICE ROUTINES ENDF TAPES	BNL	236
SCARF2, SCATTER FROM RADIATOR			FINS SNAP GEOM	AI 7090 F2 RS =	110
HERE ROA 360 F4	RS PLT=	FIRE2, 1-D AGE-DIFFUSION SLAB CYLINDER SP			9
ETRY LRLL 709 F2	RS =	FIRN, 2-D FEW-GP S4 APPROXIMATION RZ GEOM			7
NL 7090 F4	RSBP =	MUFFLE, FISSILE NUCLIDE X-SECTION EVALUATION	OR		323
P T= DAFT1, LEAST SQUARES FIT		FISSILE NUCLIDE DATA	BAPL 6600 F4 RS		R327
7044 F4	RS P =	RAD2, HTGR FISSION PRODUCT ACTIVITY DIST STUDY	GA		231
S P = FREVAP6, HTGR METALLIC		FISSION PRODUCT RELEASE	GGA 1108 F4 R		301
CINDER(MO102), POINT DEPLETION		FISSION PRODUCT	BAPL 6600 F4 RS P =		313
RS P T= DAFT1, LEAST SQUARES		FIT FISSILE NUCLIDE DATA	BAPL 6600 F4		R327
S P = FRANTIC, LEAST SQUARES		FIT SUM OF EXPONENTIALS	MIT 7090 F2 R		324
O F+ASC RS P T=	CURFIT, CURVE	FITTING EXPERIMENTAL DATA POINTS	KAPL 660	R	43
L LEAST SQUARES, GENERAL CURVE		FITTING	LASL 7094 F4 RS = LAS		62
TION GA 7044 F4	RS =	FLANGE1, SCATTERING LAW X-SECTION CALCULA			247
ION GESJ 635 F4	RS P T=	FLARE, 3-D REACTIVITY AND POWER DISTRIBUT			167
ION CDC 3600 F63	RSBP T=	FLARE, 3-D REACTIVITY AND POWER DISTRIBUT			167
BAPL 6600 F4	RS P T=	FLASH2, LOSS-OF-COOLANT ACCIDENT ANALYSIS			R295
OXIMATION XY GEOM	AGN 7090	FLOCO RSBP =	2DXY, 2-D MULTI-GP SN APPR		18
DOUBLE SN APPROX	BNWL 7090	FLOCO RSBP T=	GE-HAPO-S13, 1-D MULTI-GP		75
ULTI-GP X-SECTIONS	LASL 7090	FLOCO RS P =	ZOT, GROUP-COLLAPSING OF M		113
ENETRATION STUDY	LASL 7090	FLOCO RS P =	MCS, MONTE CARLO NEUTRON P		202

DP RS P T= LAG, ASSEMBLER FOR FLOCO2 INSTRUCTION SET	PW 1604 F+C	186
BAPL 6600 F4 RS P T= FLOT1(MO219), PWR FLOW TRANSIENT ANALYSIS		R331
DEL, MULTI-CHANNEL 2-D 2-PHASE FLOW AI 7094 F2 RS P = FLOW-MO		246
P T= TRANS-FUGUE1, TRANSIENT FLOW AND HEAT TRANSFER AI 7094 F2 RS		268
RS P = CROC90, ML-1 FLUID FLOW EXPERIMENT ANALYSIS AGN 7090 F2		154
RS P T= FLOT1(MO219), PWR FLOW TRANSIENT ANALYSIS BAPL 6600 F4		R331
W AI 7094 F2 RS P = FLOW-MODEL, MULTI-CHANNEL 2-D 2-PHASE FLO		246
90 F2 RS P = CROC90, ML-1 FLUID FLOW EXPERIMENT ANALYSIS AGN 70		154
N 7090 F2 RS P = BOUNCE, FLUX DIST IN MULTI-PIN FUEL ELEMENT AG		237
, 2-D PERTURBATION TDC OR 2DXY FLUX INPUT PW 1604 F63 RS = TDP		199
RS T= 2DXYL, 3-D MULTI-GP FLUX SYNTHESIS PROGRAM CYL PW 1604 F63		192
ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC ORNL 1604 F63 RS P =		322
ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC ORNL 360 F4 RS P =		322
PARAMETERS HETEROGENEOUS CALC FMA 7090 F2 RSBP = HERESY1, LATTICE		136
IES PW 1604 F+C DP RS P T= FMC-N/FMC-G, MC NEUTRON, GAMMA-RAY HISTOR		195
, 1-D PERTURBATION FOR AIM AND FOG CODES AI 7090 F2 RS = PERT		30
, 1-D PERTURBATION FOR AIM AND FOG CODES CDC 1604 F63 RS = PERT		30
, 1-D PERTURBATION FOR AIM AND FOG CODES BOEC 360 F4 RS P = PERT		30
PHERE AI 7090 F2 RS P T= FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER S		28
PHERE CDC 1604 F63 RS P T= FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER S		28
S GESJ 635 F4 RS P T= FORE2, FAST REACTOR EXCURSION CALCULATION		174
S ANL 360 F4 RS P T= FORE2, FAST REACTOR EXCURSION CALCULATION		174
GESJ 2000 F4 RS P T= FORE, FAST REACTOR EXCURSION CALCULATIONS		174
AI 360 F4 RSB LT= CLIP, FORM OR THREDES LIBRARY UTILITY ROUTINE		271
C AI 7090 F2 RS PLT= FORM, FAST NEUTRON SPECTRUM X-SECTION CAL		51
C CDC 1604 F63 RS PLT= FORM, FAST NEUTRON SPECTRUM X-SECTION CAL		51
N RZ PW 1604 F63 RS P T= FORTRAN TDC, 2-D MULTI-GP SN APPROXIMATIO		161
TIALS MIT 7090 F2 RS P = FRANTIC, LEAST SQUARES FIT SUM OF EXPONEN		324
LEASE GGA 1108 F4 RS P = FREVAP6, HTGR METALLIC FISSION PRODUCT RE		301
2 RS P = AIMFIRE, URANIUM FUEL CYCLE COST ANALYSIS AI 7090 F		55
7090 F2 RSBP = NPRFCCP, FUEL CYCLE COSTS PERFORMANCE DATA KE		146
44 F4 RS P T= REVISED-GAD, FUEL CYCLE ANALYSIS W/REFUELING GA 70		223
7044 F4 RS P = WAMPUM, FUEL CYCLE COSTS PERFORMANCE STUDY GA		224
7044 F4 RS P T= GARGOYLE, FUEL CYCLE ANALYSIS PARTIAL REFUEL GA		260
S T= GAFFE, EQUILIBRIUM FUEL CYCLE CALCULATION GGA 1108 F4		302
A 7044 F4 RS P T= ELBOW, FUEL ELEMENT STRESS ANALYSIS STUDY G		229
BOUNCE, FLUX DIST IN MULTI-PIN FUEL ELEMENT AGN 7090 F2 RS P =		237
= CYGRO2, STRESS ANALYSIS CYL FUEL ELEMENT BAPL 6600 F4 RS T		R266
600 F4 RS P T= FIGRO, LSBR FUEL SWELLING TEMPERATURE STUDY BAPL 6		R272
= AXFLU, HEAT TRANSFER MOLTEN FUEL TUBE BUNDLES LASL 7094 F2 RS P		182
= AXTHRM, HEAT TRANSFER SOLID FUEL TUBE BUNDLES LASL 7094 F2 RS P		183
= CORE, CORE CONFIGURATION FUEL TEMPERATURE CODE GA 7044 F4 RS		233
TION AI 7090 F2 RS = FUGUE, STEADY-STATE TEMPERATURE VOID FRAC		48
BNWL 1108 F4 RS P T= FCC4, FUNDAMENTAL MODE FAST REACTOR X-SEC CALC		306
TI-GP DIFFUSION XY RZ GEOMETRY GA 1108 F+BAL RS P = GAMBLE5, 2-D MUL		222
E STRESS ANALYSIS, 2-D BODIES GA 1108 F4 RS P = SAFE-PLANE, PLAN		252
SS ANALYSIS THIN SHELLS GA 1108 F4 RS P = SAFE-SHELL, STRE		253
TOR SHIELDING CODE CAVITY GEOM GA 1108 F4 RS P T= MUSCAT, VIEW FAC		259
GR ACCIDENT ANALYSIS DOSE CALC GA 1108 F4 RS T= GADOSE/DOSET, HT		261
SCATTERING LAW CALCULATION GA 1108 F4 RS P = GASKET, THERMAL		263
ON SPECTRUM X-SECTION CALC GA 7090 F2 RS PLT= GAM1, FAST NEUTR		33
GION RESONANCE INTEGRAL CALC GA 7090 F+FAF RSB T= ZUT, RESOLVED RE		41
REGION RESONANCE INTEGRAL CALC GA 7090 F+FAF RSB T= TUZ, UNRESOLVED		42
INE SCATTERING KERNEL CALC GA 7090 F2 RS T= SUMMIT, CRYSTALL		56
DIFFUSION BURNUP RZ GEOMETRY GA 7090 F+FAF RSBP T= DDB, 2-D FEW-GP		99
P DIFFUSION DEPLETION PROGRAM GA 7090 F2 RSB T= FEVER, 1-D FEW-G		117
INETICS WITH 2-D HEAT TRANSFER GA 7044 F4 RS T= BLOOST5, POINT-K		205
L FROM SCISRS X-SECTION TAPE GA 7044 F+MAP RSBP = UNPACK, RETRIEVA		206
O PLOTS FROM X-SECTION TAPES GA 7044 F4 RSBP T= CROSSPLOT, SC402		207
NS FROM B-W RESONANCE DATA GA 7044 F4 RS P = FASDOP, X-SECTIO		216
RE CCEF CALC FOR ANGULAR DIST GA 7044 F4 RS P = LEGCOEF3, LEGEND		217

TERVAL X-SECTION AVERAGING	GA	7044 F4	RS P =	GAVER, ENERGY IN	218
OVERLAP AND LATTICE EFFECTS	GA	7044 F+MAP	RS P T=	GAROL, RESONANCE	219
P DIFFUSION POWER DIST SEARCH	GA	7044 F4	RS P T=	GASP2, 1-D FEW-G	220
D FEW-GP DIFFUSION DEPLETION	GA	7044 F4	RS P T=	RELOAD-FEVER, 1-	221
TI-GP DIFFUSION XY RZ GEOMETRY	GA	7044 F+MAP	RSBP T=	GAMBLE4, 2-D MUL	222
L CYCLE ANALYSIS W/REFUELING	GA	7044 F4	RS P T=	REVISED-GAD, FUE	223
LE COSTS PERFORMANCE STUDY	GA	7044 F4	RS P =	WAMPUM, FUEL CYC	224
P DIFFUSION TEMP COEF CALC	GA	7044 F4	RS P T=	TEMCO, 1-D FEW-G	225
T PERFORMANCE AND PRICE STUDY	GA	7044 F+MAP	RS P T=	OPUS, POWER PLAN	226
NERATOR DESIGN CRITERIA COSTS	GA	7044 F4	RS P T=	STMGEN, STEAM GE	227
TAINMENT PRESSURE POST RUPTURE	GA	7044 F4	RS T=	PRECON, HTGR CON	228
ENT STRESS ANALYSIS STUDY	GA	7044 F4	RS P T=	ELBOW, FUEL ELEM	229
ESS ANALYSIS CF 2-D BODIES	GA	7044 F4	RS P T=	PIZZA, PLANE STR	230
ON PRODUCT ACTIVITY DIST STUDY	GA	7044 F4	RS P =	RAD2, HTGR FISSI	231
OLED CORE THERMAL DESIGN STUDY	GA	7044 F4	RS P =	GAZELLE5, GAS-CO	232
GURATION FUEL TEMPERATURE CODE	GA	7044 F4	RS =	CORE, CORE CONFI	233
ING LAW X-SECTION CALCULATION	GA	7044 F4	RS =	FLANGE1, SCATTER	247
S ANALYSIS AXISYMMETRIC LOAD	GA	7044 F4	RS P T=	SAFE-PCRS, STRES	250
ESS ANALYSIS AXISYMMETRIC LOAD	GA	7044 F4	RS P T=	SAFE-AXISYM, STR	251
YCLE ANALYSIS PARTIAL REFUEL	GA	7044 F4	RS P T=	GARGOYLE, FUEL C	260
CALC GA 1108 F4 RS T=				GADOSE/DOSET, HTGR ACCIDENT ANALYSIS DOSE	261
GGA 1108 F4 S T=				GAFFE, EQUILIBRIUM FUEL CYCLE CALCULATION	302
CALC GGA 1108 F4 RS P T=				GAF/GAR, SPECTRA AND GROUP-AVERAGED X-SEC	316
RATOR GGA 1108 F4 RS P =				GAKER, INELASTIC SCAT X-SECTION CALC MODE	289
USION GGA 1108 F4 RS P T=				GAKIN, 1-D MULTIGROUP TIME-DEPENDENT DIFF	310
C GA 7090 F2 RS PLT=				GAM1, FAST NEUTRON SPECTRUM X-SECTION CAL	33
C CDC 1604 F63 RS PLT=				GAM1, FAST NEUTRON SPECTRUM X-SECTION CAL	33
C ANL 3600 F63 RS LT=				GAM1, FAST NEUTRON SPECTRUM X-SECTION CAL	33
METRY GA 7044 F+MAP RSBP T=				GAMBLE4, 2-D MULTI-GP DIFFUSION XY RZ GEO	222
METRY GA 1108 F+BAL RS P =				GAMBLE5, 2-D MULTI-GP DIFFUSION XY RZ GEO	222
RAYS GEC 7094 F4 RS P =				GAMMA-P, PRODUCTION X-SECTIONS FOR GAMMA-	235
7090 F2 RS P =				GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY	AI 45
C 1604 F63 RS P =				GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY	CD 45
7090 F2 RS P =				GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM	AI 46
C 1604 F63 RS P =				GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM	CD 46
I 7090 F2 RS P =				CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	A 47
DC 1604 F63 RS P =				CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	C 47
RL 360 F4 RS P =				CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	S 47
P T= FMC-N/FMC-G, MC NEUTRON,				GAMMA-RAY HISTORIES PW 1604 F+CDP RS	195
M 7090 MAD RSB =				BURP4, GAMMA-RAY PHOTOFRACTION SOLID CRYSTAL	U 169
M 7090 MAD RSB =				BURP5, GAMMA-RAY PHOTOFRACTION WELL CRYSTAL	U 170
RSBPLT= NAP, NEUTRON-INDUCED				GAMMA-RAY RADIOACTIVITY IITR 7094 F4	314
TOAD, PROCESSING OF ANALYZER				GAMMA-RAY SPECTRA GGA 1108 F4 RS P T=	333
A-P, PRODUCTION X-SECTIONS FOR				GAMMA-RAYS GEC 7094 F4 RS P =	GAMM 235
EV BNWL 7090 F+FAP RS PLT=				GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 M	185
EV BNWL 1107 F4 RS PLT=				GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 M	185
ICS GGA 1108 F4 RS P =				GAPOTKIN, SPACE-INDEPENDENT REACTOR KINET	317
UEL GA 7044 F4 RS P T=				GARGOYLE, FUEL CYCLE ANALYSIS PARTIAL REF	260
CTS GA 7044 F+MAP RS P T=				GAROL, RESONANCE OVERLAP AND LATTICE EFFE	219
EQNS GGA 1108 F4 RS P =				GASA, STABILITY ANALYSIS REACTOR KINETICS	290
7044 F4 RS P =				GAZELLE5, GAS-COOLED CORE THERMAL DESIGN STUDY	GA 232
N GA 1108 F4 RS P =				GASKET, THERMAL SCATTERING LAW CALCULATIO	263
ARCH GA 7044 F4 RS P T=				GASP2, 1-D FEW-GP DIFFUSION POWER DIST SE	220
CH GGA 1108 F4 RS P T=				GASP7, 1-D BURNUP POWER DISTRIBUTION SEAR	319
G GA 7044 F4 RS P =				GAVER, ENERGY INTERVAL X-SECTION AVERAGIN	218
STUDY GA 7044 F4 RS P =				GAZELLE5, GAS-COOLED CORE THERMAL DESIGN	232
OID X-SECTION CALCULATION				GEC 625 F4 RS P =	AVOID, ANNULAR V 276
ION X-SECTIONS FOR GAMMA-RAYS				GEC 7094 F4 RS P =	GAMMA-P, PRODUCT 235
X BNWL 7090 FLOCO RSBP T=				GE-HAPO-S13, 1-D MULTI-GP DOUBLE SN APPRO	75
4 F4 RS P T=				STMGEN, STEAM GENERATOR DESIGN CRITERIA COSTS	GA 704 227
IVITY AND POWER DISTRIBUTION				GESJ 635 F4 RS P T=	FLARE, 3-D REACT 167

TOR EXCURSION CALCULATIONS	GESJ	635	F4	RS P T=	FORE2, FAST REAC	174
ATE THERMAL-HYDRAULIC ANALYSIS	GESJ	635	F4	RS P T=	MANTA, STEADY-ST	256
ESONANCE EPITHERMAL X-SECTIONS	GESJ	635	F4	RS P T=	REAX, RESOLVED R	257
F PULSED NEUTRON SOURCE DATA	GESJ	635	F4	RS P T=	EXPAN, ANALYSIS O	258
-GP DIFFUSION SYNTHESIS CALC	GESJ	635	F4	RS PLT=	BISYN, 2-D MULTI	287
OR EXCURSION CALCULATIONS	GESJ	2000	F4	RS P T=	FORE, FAST REACT	174
CE INTEGRAL X-SECTION CALC	GESJ	2000	F2	RS P =	RAPTURE, RESONAN	176
RESONANCE X-SECTION CALC	GESJ	2000	F2	RS P =	DOPIE, RESOLVED	177
Y-AVERAGED DOPPLER EFFECTS	GESJ	2000	F2	RS P =	SPARTA, SPATIAL	178
EMENT ISOTOPIC BURNUP ANALYSIS	GEV	635	F4	RS P T=	BURNUP, HEAVY EL	311
TRUM AND X-SECTION CALCULATION	GEV	7094	F+FAP	RS P T=	EPITHERMOS, SPEC	201
SCAT X-SECTION CALC MODERATOR	GGA	1108	F4	RS P =	GAKE, INELASTIC	289
ANALYSIS REACTOR KINETICS EQNS	GGA	1108	F4	RS P =	GASA, STABILITY	290
SCAT X-SECTIONS HEX LATTICE	GGA	1108	F4	RS =	HEXSCAT, ELASTIC	291
CAL RESONANCE PARAMETER CALC	GGA	1108	F4	RS P =	PSEUDO, STATISTI	292
ERATION FOR SAFE PROGRAMS	GGA	1108	F4	RS P =	GRDWRK, GRID GEN	296
-SECTIONS FAST THERMAL SPECTRA	GGA	1108	F4	RS PLT=	GGC4, MULTI-GP X	298
ELASTIC ANALYSIS CONCRETE	GGA	1108	F4	RS P =	SAFE-CREEP, VISC	300
TALLIC FISSION PRODUCT RELEASE	GGA	1108	F4	RS P =	FREVAP6, HTGR ME	301
UM FUEL CYCLE CALCULATION	GGA	1108	F4	S T=	GAFFE, EQUILIBRI	302
D KINETICS 2-D HEAT TRANSFER	GGA	1108	F4	RS P T=	BLOOST6, COMBINE	303
GROUP TIME-DEPENDENT DIFFUSION	GGA	1108	F4	RS P T=	GAKIN, 1-D MULTI	310
AND GROUP-AVERAGED X-SEC CALC	GGA	1108	F4	RS P T=	GAF/GAR, SPECTRA	316
INDEPENDENT REACTOR KINETICS	GGA	1108	F4	RS P =	GAPOTKIN, SPACE-	317
IGROUP DIFFUSION AND DEPLETION	GGA	1108	F4	RS P T=	FEVER7, 1-D MULT	318
P POWER DISTRIBUTION SEARCH	GGA	1108	F4	RS P T=	GASP7, 1-D BURNU	319
URE COEFFICIENT CALCULATION	GGA	1108	F4	RS P T=	TEMCO7, TEMPERAT	320
POSITE STRUCTURE STRESS STUDY	GGA	1108	F4	RS P T=	SAFE-3D, 3-D COM	332
OF ANALYZER GAMMA-RAY SPECTRA	GGA	1108	F4	RS P T=	TOAD, PROCESSING	333
ECTRA	GGA	1108	F4	RS PLT=	GGC4, MULTI-GP X-SECTIONS FAST THERMAL SP	298
RY	AI	7090	F2	RS P =	GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMET	45
RY	CDC	1604	F63	RS P =	GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMET	45
GEOM	AI	7090	F2	RS P =	GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE	46
GEOM	CDC	1604	F63	RS P =	GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE	46
ERS	AI	7094	F+FAP	RS PLT=	GRAVE, GROUP-AVERAGING X-SECTIONS PARAMET	162
	GGA	1108	F4	RS P =	GRDWRK, GRID GENERATION FOR SAFE PROGRAMS	296
A	1108	F4		RS P =	GRDWRK, GRID GENERATION FOR SAFE PROGRAMS	296
				RS P T=	GAF/GAR, SPECTRA AND	
I	7094	F+FAP		RS PLT=	GRAVE, GROUP-AVERAGING X-SECTIONS PARAMETERS	162
LASL	7090	FLOCO		RS P =	ZOT, GROUP-COLLAPSING OF MULTI-GP X-SECTIONS	113
RING	CDC	1604	F63	RS P =	HAFEVER, HAUSER-FESHBACH INELASTIC SCATTE	14
EM BNL-SRL	7090	F2		RS PLT=	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYST	277
EM BNL-SRL	360	F4		RS PLT=	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYST	277
CODE AGN	7090	F2		RSB =	HATCHET, COUPLED NEUTRONICS-HYDRODYNAMICS	153
	1604	F63		RS P =	HAFEVER, HAUSER-FESHBACH INELASTIC SCATTERING	14
FAP				RS P T=	RATH, 2- OR 3-D	
				RS P T=	HOT2, 2-D TRANSIENT	
ASL	7094	F2		RS P =	AXFLU, HEAT CONDUCTION LUMPED MASS	242
SL	7094	F2		RS P =	AXTHRM, HEAT CONDUCTION PROGRAM	286
ATING2, TRANSIENT STEADY-STATE					BAPL 6600 F4	
OCST5, POINT-KINETICS WITH 2-D					HEAT TRANSFER MOLTEN FUEL TUBE BUNDLES	182
ANS-FUGUE1, TRANSIENT FLOW AND					HEAT TRANSFER SOLID FUEL TUBE BUNDLES	183
BLOOST6, COMBINED KINETICS 2-D					HEAT TRANSFER AI 7094 F2	198
NSFER AI	7094	F2		RS P T=	HE	
V	635	F4		RS P T=	BL	205
					HEAT TRANSFER GA 7044 F4	
					RS T=	268
					HEAT TRANSFER AI 7094 F2	
					RS P T=	303
					HEATING2, TRANSIENT STEADY-STATE HEAT TRA	198
					HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS	311
					GE	
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					HERESY1, LATTICE PARAMETERS HETEROGENEOUS	136
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8 F4 RS T= GADOSE/DOSET,	HTGR ACCIDENT ANALYSIS DOSE CALC GA 110		261
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1108 F4 RS P = FREVAP6,	HTGR METALLIC FISSION PRODUCT RELEASE GGA		301
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7090 F+MAP RS LT= QUICKIE,	INFINITE MEDIUM SPECTRUM X-SECTIONS AI		119
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4 F4 RS P = GAVER, ENERGY	INTERVAL X-SECTION AVERAGING GA 704		218
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ORNL 7090 F2 RS P =	ISOCRUNCH, REACTION DECAY CHAIN ANALYSIS		180
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N ORNL 1604 F63 RS P T=	JUPITOR1, COUPLED-CHANNEL X-SEC EVALUATIO		308
TTING EXPERIMENTAL DATA POINTS	KAPL 6600 F+ASC RS P T= CURFIT, CURVE FI	R 43	
LYSIS SHELLS OF REVOLUTION	KAPL 6600 F4 RS P T= SOR2, STRESS ANA	R 80	
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LED PIPING SYSTEM ANALYSIS	KE 3600 F4 RSBP = WHAM, LIQUID-FIL	278	
CLE COSTS PERFORMANCE DATA	KE 7090 F2 RSBP = NPRFCCP, FUEL CY	146	
RESSURE TEMPERATURE HISTORY	KE 7094 F2 RSBP T= PTH1, BLOWDOWN P	155	
WN1, BLOWDOWN ANALYSIS LOFT	KE 7094 F+MAP RSBP T= SATURATED BLOWDO	200	
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102, SPACE-INDEPENDENT INVERSE	KINETICS CALC ANL 3600 F63 RSBP = R	168	
102, SPACE-INDEPENDENT INVERSE	KINETICS CALC WAL 7094 F4 RS = R	168	
BP = R101, SPACE-INDEPENDENT	KINETICS KEX OPTIONS ANL 3600 F63 RS	255	
P = R101, SPACE-INDEPENDENT	KINETICS KEX OPTIONS WAL 7094 F4 RS	255	
SA, STABILITY ANALYSIS REACTOR	KINETICS EQNS GGA 1108 F4 RS P = GA	290	
RS P T= BLOOST6, COMBINED	KINETICS 2-D HEAT TRANSFER GGA 1108 F4	303	
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N, SPATIALLY-DEPENDENT REACTOR	KINETICS AI 360 F4 RS P T= TS	309	
KIN, SPACE-INDEPENDENT REACTOR	KINETICS GGA 1108 F4 RS P = GAPOT	317	
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ROX SLAB CYL SPHERE PW 1604	LAG1 RS P = MGDSN, 1-D MULTI-GP SN APP	211	
PW 1604 F+CDP RS P T=	LAG, ASSEMBLER FOR FLOCO2 INSTRUCTION SET	186	
ICE WAPD 7094 F+MAP RS PLT=	LASER, SPECTRUM CALC WITH BURNUP CYL LATT	249	
GP DISCRETE ORDINATE PROGRAM	LASL 6600 F4 RS P T= DTF4, 1-D MULTI-	209	
ES, GENERAL CURVE FITTING	LASL 7094 F4 RS = LASL LEAST SQUAR	62	
PSING OF MULTI-GP X-SECTIONS	LASL 7090 FLOCO RS P = ZOT, GROUP-COLLA	113	
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NEUTRON PENETRATION STUDY	LASL 7090 FLOCO RS P = MCS, MONTE CARLO	202	
GP DISCRETE ORDINATE PROGRAM	LASL 7030 F4 RS P T= DTF4, 1-D MULTI-	209	
ATION FOR 2-D DESIGN PROGRAMS	LASL 7090 F2 RS P T= DPC, DATA PREPAR	234	
HEAT CONDUCTION LUMPED MASS	LASL 7094 FAP RS P T= RATH, 2- OR 3-D	242	

LTI-GP DTF4 WITH DEPLETION	LASL 7030 F4	RS P T= DTF-BURN, 1-D MU	269
PRODUCED WIRING LISTS UHTREX	LASL 7090 F2	RS P = WIREX, COMPUTER-	315
LASL 7094 F4 RS =	LASL LEAST SQUARES, GENERAL CURVE FITTING		62
7090 F2 RSBP = HERESY1,	LATTICE PARAMETERS HETEROGENEOUS CALC FMA		136
GAROL, RESONANCE OVERLAP AND	LATTICE EFFECTS GA 7044 F+MAP RS P T=		219
SPECTRUM CALC WITH BURNUP CYL	LATTICE WAPD 7094 F+MAP RS PLT= LASER,		249
T, ELASTIC SCAT X-SECTIONS HEX	LATTICE GGA 1108 F4 RS = HEXSCA		291
1A, DOSE RATE FROM SNAP SHIELD	LEAKAGE AI 7090 F2 RS = FARSE		91
LASL 7094 F4 RS = LASL	LEAST SQUARES, GENERAL CURVE FITTING		62
L 7094 F2 RS P = EXPALS,	LEAST SQUARES EXPONENTIAL DECAY CURVES LR		321
7090 F2 RS P = FRANTIC,	LEAST SQUARES FIT SUM OF EXPONENTIALS MIT		324
APL 6600 F4 RS P T= DAFT1,	LEAST SQUARES FIT FISSILE NUCLIDE DATA B		R327
DIST GA 7044 F4 RS P =	LEGCDEF3, LEGENDRE COEF CALC FOR ANGULAR		217
AI 7094 F+MAP RS P = CHAD,	LEGENDRE COEF CALC FOR ANGULAR DIST DATA		215
7044 F4 RS P = LEGCOEF3,	LEGENDRE COEF CALC FOR ANGULAR DIST GA		217
ON WAPD 7090 F+FAP RSBPLT=	LEOPARD, SPECTRA CALCULATION WITH DEPLETI		279
BLST, 1-D MULTI-GP DIFFUSION +	LIB AI 360 F4 RSBPLT= CAESAR4/LI		270
+FAP RS LT= MOMUS, X-SECTION	LIBRARY UTILITY PROGRAM AI 7094 F		159
3 RSBP T= XLIBIT, X-SECTION	LIBRARY UTILITY ROUTINE ANL 3600 F6		181
RS = CSP2A, SN X-SECTION	LIBRARY TAPE PREPARATION PW 1604 F63		193
RS = CSP1, SN X-SECTION	LIBRARY TAPE PREPARATION PW 1604 F63		194
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P T= MO552, DYNAMIC ANALYSIS	LINEAR ELASTIC SYSTEMS BAPL 6600 F4 RS		R283
M KAPL 6600 F4 RS P T=	LION, 3-D TEMPERATURE DISTRIBUTION PROGRA		R299
TION DAC 7090 F2 RSBP =	LIPRECAN1, MC NEUTRON PENETRATION CALCULA		123
KE 3600 F4 RSBP = WHAM,	LIQUID-FILLED PIPING SYSTEM ANALYSIS		278
0 F4 RS P = WATER-HAMMER,	LIQUID BLOWDOWN ANALYSIS LOFT UG 36		278
, STRESS ANALYSIS AXISYMMETRIC	LOAD GA 7044 F4 RS P T= SAFE-PCRS		250
, SHELL STRESS ANALYSIS AXISYM	LOAD BAPL 6600 F4 RS P T= SEALSHLL2		R282
, STRESS ANALYSIS AXISYMMETRIC	LOAD GA 7044 F4 RS P T= SAFE-AXISYM		251
D BLOWDOWN1, BLOWDOWN ANALYSIS	LOFT KE 7094 F+MAP RSBP T= SATURATE		200
MMER, LIQUID BLOWDOWN ANALYSIS	LOFT UG 360 F4 RS P = WATER-HA		278
00 F4 RS P T= MO555(ACT1),	LOSS-OF-COOLANT ACCIDENT ANALYSIS BAPL 66		R284
PL 6600 F4 RS P T= FLASH2,	LOSS-OF-COOLANT ACCIDENT ANALYSIS BA		R295
7040 F+MAP RS P T= CONTEMPT,	LOSS-OF-COOLANT ACCIDENT ANALYSIS PPC		297
400 F4 RS P T= NURLOC-1.0,	LOSS-OF-COOLANT THERMAL ANALYSIS BMI 6		328
I 6400 F4 RS P T= ECCSA1,	LOSS-OF-COOLANT + EMERGENCY COOLING BM		330
GP DISCRETE ORDINATE PROGRAM	LRC 7094 F4 RS P T= DTF4, 1-D MULTI-		209
ROUP DISCRETE ORDINATE PROGRAM	LRC 7090 F+MAP RS P T= TDSN, 2-D MULTIG		312
UARES EXPONENTIAL DECAY CURVES	LRL 7094 F2 RS P = EXPALS, LEAST SQ		321
S4 APPROXIMATION RZ GEOMETRY	LRL 709 F2 RS = FIRN, 2-D FEW-GP		7
EUTRONICS-HYDRODYNAMICS SPHERE	LRL 7090 F2 RS P = CONEC, COUPLED N		129
LTI-GP TRANSFER COEFFICIENTS	LRL 7090 F+FAP RS P T= SOPHIST1/2/5, MU		160
APL 6600 F4 RS P T= FIGRO,	LSBR FUEL SWELLING TEMPERATURE STUDY B		R272
HERE ANL 3600 F63 RS P T=	MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SP		262
RY BNWL 7090 F2 RSBPLT=	MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMET		143
RY ANL 3600 F63 RSBPLT=	MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMET		143
POINT SOURCE UM 7090	MAD RSB = BURP1, DETECTOR EFFICIENCY		164
DISK SOURCE UM 7090	MAD RSB = BURP2, DETECTOR EFFICIENCY		165
POINT SOURCE UM 7090	MAD RSB = BURP3, DETECTOR EFFICIENCY		166
TION SOLID CRYSTAL UM 7090	MAD RSB = BURP4, GAMMA-RAY PHOTOFRAC		169
TION WELL CRYSTAL UM 7090	MAD RSB = BURP5, GAMMA-RAY PHOTOFRAC		170
LYSIS GESJ 635 F4 RS P T=	MANTA, STEADY-STATE THERMAL-HYDRAULIC ANA		256
TRY APDA 7094 F4 RS P =	MARS, 2-D EXCURSION CALCULATION R-Z GEOME		293
OR 3-D HEAT CONDUCTION LUMPED	MASS LASL 7094 FAP RS P T= RATH, 2-		242
7090 F2 RSBP = LIPRECAN1,	MC NEUTRON PENETRATION CALCULATION DAC		123
04 F+CDP RS P T= FMC-N/FMC-G,	MC NEUTRON, GAMMA-RAY HISTORIES PW 16		195
Y LASL 7090 FLOCO RS P =	MCS, MONTE CARLO NEUTRON PENETRATION STUD		202
8 F4 RS P = FREVAP6, HTGR	METALLIC FISSION PRODUCT RELEASE GGA 110		301
HERE PW 1604 LAG1 RS P =	MGDSN, 1-D MULTI-GP SN APPROX SLAB CYL SP		211
NEOUS ANL 3600 F63 RSBP =	MISH-MASH, RESONANCE INTEGRAL CALC HOMOG		214

GEOM PP	7090 F2	RS P =	MIST, 1-D FEW-GP SN DOUBLE SN APPROX SLAB	59
QUARES FIT SUM OF EXPONENTIALS	MIT 7090 F2	RS P =	FRANTIC, LEAST S	324
N 7090 F2	RS P =	CROC90, ML-1 FLUID FLOW EXPERIMENT ANALYSIS	AG	154
090 F2	RS	=	DANCOFF JR, MODERATOR SPACE CHORD DIST FUNCT	AEG 7 150
INELASTIC SCAT X-SECTION CALC	MODERATOR GGA 1108 F4	RS P =	GAKEP,	289
RS P =	AXFLU, HEAT TRANSFER	MOLTEN FUEL TUBE BUNDLES	LASL 7094 F2	182
RESTRAINT PIPE STRESS, MAXIMUM	MOMENT CALC	AI	7090 F+FAP RS T= 4	109
AI	7094 F+FAP RS LT=	MOMUS, X-SECTION LIBRARY UTILITY PROGRAM		159
7094 F+FAP RS PLT=	TYCHE3, MONTE CARLO SLOWING-DOWN DENSITY CALC	AI		149
LASL 7090 FLOCO RS P =	MCS, MONTE CARLO NEUTRON PENETRATION STUDY			202
F4 RS P T=	HWOGR-SAFE, 2-D MONTE CARLO CELL CALCULATION	AI	360	307
ETRY AI	7090 F2 RS	=	MORTIMER, DOSE RATE CALCULATION SNAP GEOM	142
TEM BAPL	6600 F4 RS P T=	M0457(PIPE), ELASTIC STRESS OF PIPING SYS	R329	
STEMS BAPL	6600 F4 RS P T=	M0552, DYNAMIC ANALYSIS LINEAR ELASTIC SY	R283	
YSIS BAPL	6600 F4 RS P T=	M0555(ACT1), LOSS-OF-COOLANT ACCIDENT ANA	R284	
BAPL	6600 F4 RS P T=	M0899/HOH, STEAM TABLES 14.5-2538 PSIA	R294	
-SECS BAPL	6600 F4 RS	T=	M0807, 2-D DIFFUSION ABSORPTION REMOVAL X	R280
ION ORNL	7090 F4 RSBP	=	MUFFLE, FISSILE NUCLIDE X-SECTION EVALUAT	323
090 F2 RS	LT=	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	AI 7	58
604 F63 RS	LT=	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	CDC 1	58
360 F4 RSBP	T=	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	AI	58
108 F4 RS	P T=	FEVER7, 1-D MULTIGROUP DIFFUSION AND DEPLETION	GGA 1	318
L 1108 F4 RS	P T=	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	BNW	325
1604 F63 RS	PLT=	DTF, 1-D MULTIGROUP DISCRETE ORDINATE CALC	UNC	144
7090 F+MAP RS	P T=	TDSN, 2-D MULTIGROUP DISCRETE ORDINATE PROGRAM	LRC	312
RSBPLT=	MAC, SHIELD DESIGN	MULTIGROUP SLAB GEOMETRY	BNWL 7090 F2	143
RSBPLT=	MAC, SHIELD DESIGN	MULTIGROUP SLAB GEOMETRY	ANL 3600 F63	143
1108 F4 RS	P T=	GAKIN, 1-D MULTIGROUP TIME-DEPENDENT DIFFUSION	GGA	310
094 F2 RS	P	=	FLOW-MODEL, MULTI-CHANNEL 2-D 2-PHASE FLOW	AI 7 246
L 7090 F+FAP RS	PLT=	GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV	BNW	185
L 1107 F4 RS	PLT=	GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV	BNW	185
SBPLT=	AGN-GAM, FAST SPECTRUM	MULTI-GP CONSTANT CALC	AGN 7090 F2 R	204
7090 F+FAP RS	PLT=	AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	AI	29
1604 F63 RS	PLT=	AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	CDC	29
FAP RSBPLT=	CRAM, 1-D AND 2-D	MULTI-GP DIFFUSION PROGRAM	UK-R 7090 F+	103
7090 F+FAP RS	LT=	ULCER, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	AI	118
7090 F+FAP RS	LT=	FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	AI	120
1604 F63 RS	PLT=	FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	CDC	120
360 F4 RS	P T=	FAIMOS, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	BOEC	120
FAP RS	P T=	EXTERMINATOR, 2-D MULTI-GP DIFFUSION PROGRAM	ORNL 7090 F+	156
RS	P T=	EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PROGRAM	ORNL 360 F4	156
P T=	VARI-QUIR, TIME-DEP	2-D MULTI-GP DIFFUSION	WAL 7090 F+FAP RS	212
44 F+MAP RSBP	T=	GAMBLE4, 2-D MULTI-GP DIFFUSION XY RZ GEOMETRY	GA 70	222
08 F+BAL RS	P	=	GAMBLE5, 2-D MULTI-GP DIFFUSION XY RZ GEOMETRY	GA 11 222
90 F+FAP RS	P T=	ASSAULT, 2-D MULTI-GP DIFFUSION DEPLETION CODE	ORNL 70	240
L 1107 F4 RS	P T=	HFN, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	BNW	241
7090 F2 RS	PLT=	DTF2, 1-D MULTI-GP DISCRETE ORDINATE CALC	AI	151
3600 F63 RS	P T=	MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	ANL	262
F4 RS	P T=	VARI-QUIR3, 2-D MULTI-GP DIFFUSION XY RZ RTH	WAL 7094	264
RSBPLT=	CAESAR4/LIBLST, 1-D	MULTI-GP DIFFUSION + LIB	AI 360 F4	270
635 F4 RS	PLT=	BISYN, 2-D MULTI-GP DIFFUSION SYNTHESIS CALC	GESJ	287
360 F4 RSBPLT=	ANISN, 1-D	MULTI-GP DISCRETE ORDINATE CALC	AI	151
L 1604 F63 RS	=	2DF, 2-D MULTI-GP DISCRETE ORDINATE CODE	UNC-LAS	173
7030 F4 RS	P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	LASL	209
7094 F4 RS	P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	LRC	209
6600 F4 RS	P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	LASL	209
625 F4 RS	P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	BEC	209
360 F4 RS	P T=	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	ANL	209
O F36 RS	P T=	SNARG-1D, 1-D MULTI-GP DISCRETE ORDINATE CALC	ANL 360	288
LOCO RSBP	T=	GE-HAPO-S13, 1-D MULTI-GP DOUBLE SN APPROX	BNWL 7090 F	75
O F4 RS	P T=	DTF-BURN, 1-D MULTI-GP DTF4 WITH DEPLETION	LASL 703	269

1604 F63	RS	T= 2DXYL, 3-D	MULTI-GP FLUX SYNTHESIS PROGRAM CYL PW	192
7090 F2	RS P T=	W-DSN, 1-D	MULTI-GP SN APPROX SLAB CYL SPHERE UK-W	132
1604 LAG1	RS P =	MGDSN, 1-D	MULTI-GP SN APPROX SLAB CYL SPHERE PW	211
7090 FLOCO	RSBP =	2DXY, 2-D	MULTI-GP SN APPROXIMATION XY GEOM AGN	18
63	RS P T=	FORTTRAN TDC, 2-D	MULTI-GP SN APPROXIMATION RZ PW 1604 F	161
0 F+FAP	RS P T=	SOPHIST1/2/5,	MULTI-GP TRANSFER COEFFICIENTS LRLL 709	160
	RS P T=	AGN-SIGMA, CALC OF	MULTI-GP TRANSFER MATRICES AGN 7090 F2	243
P = ZOT,	GROUP-COLLAPSING OF	MULTI-GP X-SECTIONS	LASL 7090 FLOCO RS	113
GGA 1108 F4	RS PLT=	GGC4,	MULTI-GP X-SECTIONS FAST THERMAL SPECTRA	298
RS P =	COMBCO, COMBINED B-W	MULTI-LEVEL CONVOLUTION	PP 7040 F4	203
RS P =	BOUNCE, FLUX DIST IN	MULTI-PIN FUEL ELEMENT	AGN 7090 F2	237
GEOM GA 1108 F4	RS P T=	MUSCAT, VIEW FACTOR SHIELDING CODE	CAVITY	259
ITY. IITR 7094 F4	RSBPLT=	NAP, NEUTRON-INDUCED GAMMA-RAY RADIOACTIV		314
ANL 3600 F63	RSBP =	NEARREX, COMPOUND NUCLEUS X-SECTION CALC		171
LOCO RS P =	MCS, MONTE CARLO	NEUTRON PENETRATION STUDY	LASL 7090 F	202
1604 F63	RS PLT=	GAM1, FAST	NEUTRON SPECTRUM X-SECTION CALC CDC	33
F+CDP RS P T=	FMC-N/FMC-G, MC	NEUTRON, GAMMA-RAY HISTORIES	PW 1604	195
P =	WEAK EXPLOSION, COUPLED	NEUTRON-HYDRODYNAMICS	APDA 7094 F2 RS	145
IITR 7094 F4	RSBPLT=	NAP, NEUTRON-INDUCED GAMMA-RAY RADIOACTIVITY		314
2	RSB =	HATCHET, COUPLED	NEUTRONICS-HYDRODYNAMICS CODE AGN 7090 F	153
00 F63	RSBP =	AX1, COUPLED	NEUTRONICS-HYDRODYNAMICS SPHERE CDC 36	102
F2	RS P =	CONEC, COUPLED	NEUTRONICS-HYDRODYNAMICS SPHERE LRLL 7090	129
F63	RS P =	AX-TNT, COUPLED	NEUTRONICS-HYDRODYNAMICS SPH PW 1604	191
I 7094 F2	RS P T=	2PLUS,	NON-SPHERICAL OPTICAL MODEL X-SECTIONS A	254
NL 3600 F63	RSBP T=	2PLUS,	NON-SPHERICAL OPTICAL MODEL X-SECTIONS A	254
A	KE 7090 F2	RSBP =	NRRFCCP, FUEL CYCLE COSTS PERFORMANCE DAT	146
	RSBP =	NUCY, SOLUTION OF	NUCLIDE CHAIN EQUATIONS ORNL 7090 F2	134
3	RSBP =	NUCY, SOLUTION OF	NUCLIDE CHAIN EQUATIONS ORNL 1604 F6	134
FT1, LEAST SQUARES FIT FISSILE	NUCLIDE DATA	BAPL 6600 F4	RS P T= DA	R327
F4	RSBP =	MUFFLE, FISSILE	NUCLIDE X-SECTION EVALUATION ORNL 7090	323
	ORNL 7090 F2	RSBP =	NUCY, SOLUTION OF NUCLIDE CHAIN EQUATIONS	134
	ORNL 1604 F63	RSBP =	NUCY, SOLUTION OF NUCLIDE CHAIN EQUATIONS	134
SIS BMI 6400 F4	RS P T=	NURLOC-1.0, LOSS-OF-COOLANT THERMAL ANALY		328
RS P T=	2PLUS, NON-SPHERICAL	OPTICAL MODEL X-SECTIONS	AI 7094 F2	254
RSBP T=	2PLUS, NON-SPHERICAL	OPTICAL MODEL X-SECTIONS	ANL 3600 F63	254
ROCK, SPACE POWER PLANT DESIGN	OPTIMIZATION	AI 7090 F2	RS = C	112
HOCK, SPACE POWER PLANT DESIGN	OPTIMIZATION	AI 7090 F2	RS = S	114
RS P =	SHOE, SHIELD WEIGHT	OPTIMIZATION DOSE CALC	AI 7094 F2	197
=	WOPEXPT, REACTOR WEIGHT	OPTIMIZATION STUDY	PW 1604 F63 RS	190
TUDY GA 7044 F+MAP	RS P T=	OPUS, POWER PLANT PERFORMANCE AND PRICE S		226
-D MULTI-GP DIFFUSION PROGRAM	ORNL 360 F4	RS P T=	EXTERMINATOR2, 2	156
PE PRODUCTION FLUX, X-SEC CALC	ORNL 360 F4	RS P =	ISOSEARCH, ISOTO	322
F NUCLIDE CHAIN EQUATIONS	ORNL 1604 F63	RSBP =	NUCY, SOLUTION O	134
M YIELD FROM REACTION OR DECAY	ORNL 1604 F63	RS P =	ISOTOPES, MAXIMU	179
ION DECAY CHAIN ANALYSIS	ORNL 1604 F63	RS P =	ISOCRUNCH, REACT	180
D-CHANNEL X-SEC EVALUATION	ORNL 1604 F63	RS P T=	JUPITOR1, COUPLE	308
PE PRODUCTION FLUX, X-SEC CALC	ORNL 1604 F63	RS P =	ISOSEARCH, ISOTO	322
CATTERING PHASE-SHIFT ANALYSIS	ORNL 1604 F63	RS P =	PEGGY, ELASTIC S	334
WAVE FUNCTION RADIAL INTEGRALS	ORNL 1604 F63	RS P =	RAMES, PARTICLE	335
-GROUP DIFFUSION XYZ GEOMETRY	ORNL 7090 F2	RS P T=	WHIRLAWAY, 3-D 2	32
2-GROUP DIFFUSION SLAB CYL	ORNL 7090 F2	RS P T=	EQUIPOISE3, 2-D	39
-GROUP DIFFUSION SLAB CYLINDER	ORNL 7090 F2	RS P T=	20GRAND, 2-D FEW	40
2-GP DIFFUSION CYLINDER SLAB	ORNL 7090 F2	RS P T=	EQUIPOISE3A, 2-D	87
F NUCLIDE CHAIN EQUATIONS	ORNL 7090 F2	RSBP =	NUCY, SOLUTION O	134
D MULTI-GP DIFFUSION PROGRAM	ORNL 7090 F+MAP	RS P T=	EXTERMINATOR, 2-	156
ION DECAY CHAIN ANALYSIS	ORNL 7090 F2	RS P =	ISOCRUNCH, REACT	180
TI-GP DIFFUSION DEPLETION CODE	ORNL 7090 F+MAP	RS P T=	ASSAULT, 2-D MUL	240
NUCLIDE X-SECTION EVALUATION	ORNL 7090 F4	RSBP =	MUFFLE, FISSILE	323
7090 F2	RS =	WED, W-DSN	OUTPUT TAPE EDIT REACTION RATES UK-W	133
EFFECTIVE X-SECTION CALC FROM DSN	OUTPUT	PW 1604 LAG1	RS P = DTX, EFF	210
+MAP RS P T=	GAROL, RESONANCE	OVERLAP AND LATTICE EFFECTS	GA 7044 F	219

RNL 1604 F63	RS P = RAMES,	PARTICLE WAVE FUNCTION RADIAL INTEGRALS O	335
ON IBM 360 F+BAL	RS P T=	PDQ5, 2-D FEW-GROUP DIFFUSION AND DEPLETI	R336
ON BAPL 6600 F+ASC	RS T=	PDQ7, 1,2 OR 3-D FEW-GP DIFFUSION DEPLETI	R275
YSIS ORNL 1604 F63	RS P =	PEGGY, ELASTIC SCATTERING PHASE-SHIFT ANA	334
RSBP = LIPRECAN1, MC NEUTRON		PENETRATION CALCULATION DAC 7090 F2	123
P = MCS, MONTE CARLO NEUTRON		PENETRATION STUDY LASL 7090 FLOCO RS	202
MAP RS P T= OPUS, POWER PLANT		PERFORMANCE AND PRICE STUDY GA 7044 F+	226
= NPRFCCP, FUEL CYCLE COSTS		PERFORMANCE DATA KE 7090 F2 RSBP	146
P = WAMPUM, FUEL CYCLE COSTS		PERFORMANCE STUDY GA 7044 F4 RS	224
ETRY BNWL 1108 F4	RS P =	PERT4, 2-D PERTURBATION XY RZ RTHETA GEOM	304
7090 F2	RS = PERT, 1-D	PERTURBATION FOR AIM AND FOG CODES AI	30
1604 F63	RS = PERT, 1-D	PERTURBATION FOR AIM AND FOG CODES CDC	30
360 F4	RS P = PERT, 1-D	PERTURBATION FOR AIM AND FOG CODES BOEC	30
1604 F63	RS = TDP, 2-D	PERTURBATION TDC OR 2DXY FLUX INPUT PW	199
1108 F4	RS P = PERT4, 2-D	PERTURBATION XY RZ RTHETA GEOMETRY BNWL	304
DES AI 7090 F2	RS =	PERT, 1-D PERTURBATION FOR AIM AND FOG CO	30
DES CDC 1604 F63	RS =	PERT, 1-D PERTURBATION FOR AIM AND FOG CO	30
DES BOEC 360 F4	RS P =	PERT, 1-D PERTURBATION FOR AIM AND FOG CO	30
= PEGGY, ELASTIC SCATTERING		PHASE-SHIFT ANALYSIS ORNL 1604 F63 RS P	334
AD RSB = BURP4, GAMMA-RAY		PHOTOFRACTION SOLID CRYSTAL UM 7090 M	169
AD RSB = BURP5, GAMMA-RAY		PHOTOFRACTION WELL CRYSTAL UM 7090 M	170
7090 F+MAP RS T= 4RESTRAINT		PIPE STRESS, MAXIMUM MOMENT CALC AI	109
RSBP = WHAM, LIQUID-FILLED		PIPING SYSTEM ANALYSIS KE 3600 F4	278
MO457(PIPE), ELASTIC STRESS OF		PIPING SYSTEM BAPL 6600 F4 RS P T=	R329
DY PW 1604 F63	RS P T=	PIP, CENTRIFUGAL PUMP IMPELLER DESIGN STU	187
S GA 7044 F4	RS P T=	PIZZA, PLANE STRESS ANALYSIS OF 2-D BODIE	230
A 7044 F4	RS P T= PIZZA,	PLANE STRESS ANALYSIS OF 2-D BODIES G	230
108 F4	RS P = SAFE-PLANE,	PLANE STRESS ANALYSIS, 2-D BODIES GA 1	252
RS = CROCK, SPACE POWER		PLANT DESIGN OPTIMIZATION AI 7090 F2	112
RS = SHOCK, SPACE POWER		PLANT DESIGN OPTIMIZATION AI 7090 F2	114
044 F+MAP RS P T= OPUS, POWER		PLANT PERFORMANCE AND PRICE STUDY GA 7	226
RSBP T= CROSSPLOT, SC4020		PLOTS FROM X-SECTION TAPES GA 7044 F4	207
094 F2	RS T= CPS, SC4020	PLOTS FROM SCISRS X-SECTION TAPES WAL 7	239
= BURP1, DETECTOR EFFICIENCY		POINT SOURCE UM 7090 MAD RSB	164
= BURP3, DETECTOR EFFICIENCY		POINT SOURCE UM 7090 MAD RSB	166
F4 RS P = CINDER(M0102),		POINT DEPLETION FISSION PRODUCT BAPL 6600	313
7044 F4	RS T= BLOOST5,	POINT-KINETICS WITH 2-D HEAT TRANSFER GA	205
90 F2	RS = CROCK, SPACE	POWER PLANT DESIGN OPTIMIZATION AI 70	112
90 F2	RS = SHOCK, SPACE	POWER PLANT DESIGN OPTIMIZATION AI 70	114
T= FLARE, 3-D REACTIVITY AND		POWER DISTRIBUTION GESJ 635 F4 RS P	167
T= FLARE, 3-D REACTIVITY AND		POWER DISTRIBUTION CDC 3600 F63 RSBP	167
= GASP2, 1-D FEW-GP DIFFUSION		POWER DIST SEARCH GA 7044 F4 RS P T	220
GA 7044 F+MAP RS P T= OPUS,		POWER PLANT PERFORMANCE AND PRICE STUDY	226
RS P T= GASP7, 1-D BURNUP		POWER DISTRIBUTION SEARCH GGA 1108 F4	319
P SN APPROXIMATION CYLINDER		PP 7040 F4 RS P T= TOPIC, 1-D FEW-G	148
B-W MULTI-LEVEL CONVOLUTION		PP 7040 F4 RS P = COMBCO, COMBINED	203
AL SAFETY ANALYSIS PROGRAM		PP 7040 F+MAP RS P T= RSAC, RADIOLOGIC	265
F-COOLANT ACCIDENT ANALYSIS		PP 7040 F+MAP RS P T= CONTEMPT, LOSS-O	297
SN DOUBLE SN APPROX SLAB GEOM		PP 7090 F2 RS P = MIST, 1-D FEW-GP	59
P SN APPROXIMATION CYLINDER		PP 7090 F2 RS P T= TOPIC, 1-D FEW-G	148
PTURE GA 7044 F4	RS T=	PRECON, HTGR CONTAINMENT PRESSURE POST RU	228
F2 RSBP T= PTH1, BLOWDOWN		PRESSURE TEMPERATURE HISTORY KE 7094	155
T= PRECON, HTGR CONTAINMENT		PRESSURE POST RUPTURE GA 7044 F4 RS	228
S, POWER PLANT PERFORMANCE AND		PRICE STUDY GA 7044 F+MAP RS P T= OPU	226
RS P = ISOSEARCH, ISOTOPE		PRODUCTION FLUX, X-SEC CALC ORNL 1604 F63	322
RS P = ISOSEARCH, ISOTOPE		PRODUCTION FLUX, X-SEC CALC ORNL 360 F4	322
7094 F4	RS P = GAMMA-P,	PRDUCTION X-SECTIONS FOR GAMMA-RAYS GEC	235
ALC GGA 1108 F4	RS P =	PSEUDO, STATISTICAL RESONANCE PARAMETER C	292
RY KE 7094 F2	RSBP T=	PTH1, BLOWDOWN PRESSURE TEMPERATURE HISTO	155
RS P T= EXPN, ANALYSIS OF		PULSED NEUTRON SOURCE DATA GESJ 635 F4	258
63 RS P T= PIP, CENTRIFUGAL		PUMP IMPELLER DESIGN STUDY PW 1604 F	187

MULTI-GP SN APPROXIMATION RZ	PW	1604 F63	RS P T=	FORTTRAN TDC, 2-D	161
OR FLOCO2 INSTRUCTION SET	PW	1604 F+CDP	RS P T=	LAG, ASSEMBLER F	186
PUMP IMPELLER DESIGN STUDY	PW	1604 F63	RS P T=	PIP, CENTRIFUGAL	187
R FUNCTION EVALUATION	PW	1604 F63	RS P =	CMPXMAT, TRANSFE	188
OF SN CONSTANTS FOR DSN TDC	PW	1604 F63	RS P =	SNC, CALCULATION	189
R WEIGHT OPTIMIZATION STUDY	PW	1604 F63	RS =	WOPEXPRT, REACTO	190
NEUTRONICS-HYDRODYNAMICS SPH	PW	1604 F63	RS P =	AX-TNT, COUPLED	191
-GP FLUX SYNTHESIS PROGRAM CYL	PW	1604 F63	RS T=	2DXYL, 3-D MULTI	192
ION LIBRARY TAPE PREPARATION	PW	1604 F63	RS =	CSP2A, SN X-SECT	193
ON LIBRARY TAPE PREPARATION	PW	1604 F63	RS =	CSP1, SN X-SECTI	194
NEUTRON, GAMMA-RAY HISTORIES	PW	1604 F+CDP	RS P T=	FMC-N/FMC-G, MC	195
ATION TDC OR 2DXY FLUX INPUT	PW	1604 F63	RS =	TDP, 2-D PERTURB	199
-SECTION CALC FROM DSN OUTPUT	PW	1604 LAG1	RS P =	DTX, EFFECTIVE X	210
-GP SN APPROX SLAB CYL SPHERE	PW	1604 LAG1	RS P =	MGDSN, 1-D MULTI	211
O F4 RS P T= FLOT1(MO219),	PWR	FLOW TRANSIENT ANALYSIS	BAPL 660	R331	
ONS AI 7090 F+FAP RS LT=	QUICKIE, INFINITE MEDIUM SPECTRUM X-SECTI			119	
ANL 3600 F63 RS P T=	R153, PARAMETRIC SITE REQUIREMENT STUDY			172	
ONS ANL 3600 F63 RSBP =	R101, SPACE-INDEPENDENT KINETICS KEX OPTI			255	
ONS WAL 7094 F4 RS P =	R101, SPACE-INDEPENDENT KINETICS KEX OPTI			255	
CALC ANL 3600 F63 RSBP =	R102, SPACE-INDEPENDENT INVERSE KINETICS			168	
CALC WAL 7094 F4 RS =	R102, SPACE-INDEPENDENT INVERSE KINETICS			168	
ELL ANL 3600 F36 RSBP T=	RABBLE/WLIB/FLAT, RESONANCE ABSORPTION, C			281	
STUDY GA 7044 F4 RS P =	RAD2, HTGR FISSION PRODUCT ACTIVITY DIST			231	
RAMES, PARTICLE WAVE FUNCTION	RADIAL INTEGRALS ORNL 1604 F63 RS P =			335	
RS = SCARF2, SCATTER FROM	RADIATOR FINS SNAP GEOM AI 7090 F2			110	
NAP, NEUTRON-INDUCED GAMMA-RAY	RADIOACTIVITY IITR 7094 F4 RSBPLT=			314	
PP 7040 F+MAP RS P T= RSAC,	RADIOLOGICAL SAFETY ANALYSIS PROGRAM			265	
GRALS ORNL 1604 F63 RS P =	RAMES, PARTICLE WAVE FUNCTION RADIAL INTE			335	
C GESJ 2000 F2 RS P =	RAPTURE, RESONANCE INTEGRAL X-SECTION CAL			176	
SS LASL 7094 FAP RS P T=	RATH, 2- OR 3-D HEAT CONDUCTION LUMPED MA			242	
RY AI 7090 F2 RS P =	RATRAP, DOSE RATE CALCULATION SNAP GEOMET			141	
7090 F2 RS P = ISOCRUNCH,	REACTION DECAY CHAIN ANALYSIS ORNL			180	
1604 F63 RS P = ISOCRUNCH,	REACTION DECAY CHAIN ANALYSIS ORNL			180	
ISOTOPES, MAXIMUM YIELD FROM	REACTION OR DECAY ORNL 1604 F63 RS P =			179	
= WED, W-DSN OUTPUT TAPE EDIT	REACTION RATES UK-W 7090 F2 RS			133	
635 F4 RS P T= FLARE, 3-D	REACTIVITY AND POWER DISTRIBUTION GESJ			167	
3600 F63 RSBP T= FLARE, 3-D	REACTIVITY AND POWER DISTRIBUTION CDC			167	
TIONS GESJ 635 F4 RS P T=	REAX, RESOLVED RESONANCE EPITHERMAL X-SEC			257	
E, FUEL CYCLE ANALYSIS PARTIAL	REFUEL GA 7044 F4 RS P T= GARGOYL			260	
IGN GA 7044 F4 RS P T=	RELOAD-FEVER, 1-D FEW-GP DIFFUSION DEPLET			221	
O807, 2-D DIFFUSION ABSORPTION	REMOVAL X-SECS BAPL 6600 F4 RS T= M			R280	
P T= AISITE2, PARAMETRIC SITE	REQUIREMENT STUDY AI 7094 F+FAP RS			172	
P T= AISITE2, PARAMETRIC SITE	REQUIREMENT STUDY AI 360 F4 RS			172	
RS P T= R153, PARAMETRIC SITE	REQUIREMENT STUDY ANL 3600 F63			172	
GA 7090 F+FAP RSB T= ZUT,	RESOLVED REGION RESONANCE INTEGRAL CALC			41	
ESJ 2000 F2 RS P = DOPIE,	RESOLVED RESONANCE X-SECTION CALC G			177	
GESJ 635 F4 RS P T= REAX,	RESOLVED RESONANCE EPITHERMAL X-SECTIONS			257	
W 2000 F4 RS = STRIP,	RESOLVED RESONANCE INTEGRAL CALCULATION B			305	
6 RSBP T= RABBLE/WLIB/FLAT,	RESONANCE ABSORPTION, CELL ANL 3600 F3			281	
= FASDOP, X-SECTIONS FROM B-W	RESONANCE DATA GA 7044 F4 RS P			216	
F4 RS P T= REAX, RESOLVED	RESONANCE EPITHERMAL X-SECTIONS GESJ 635			257	
RSB T= ZUT, RESOLVED REGION	RESONANCE INTEGRAL CALC GA 7090 F+FAP			41	
SB T= TUZ, UNRESOLVED REGION	RESONANCE INTEGRAL CALC GA 7090 F+FAP R			42	
I 7090 F2 RS PLT= ARES2,	RESONANCE INTEGRAL X-SECTION CALC A			89	
DC 1604 F63 RS PLT= ARES2,	RESONANCE INTEGRAL X-SECTION CALC C			89	
J 2000 F2 RS P = RAPTURE,	RESONANCE INTEGRAL X-SECTION CALC GES			176	
I 7094 F+MAP RS PLT= TRIX1,	RESONANCE INTEGRAL X-SECTION CALC A			208	
3600 F63 RSBP T= RIFF-RAFF,	RESONANCE INTEGRAL CALC 2-REG CELL ANL			213	
3600 F63 RSBP = MISH-MASH,	RESONANCE INTEGRAL CALC HOMOGENEOUS ANL			214	
RS P T= RESQ2/RESQ0/DBF1,	RESONANCE INTEGRAL HEX CELL BAPL 6600 F4			R285	
F4 RS = STRIP, RESOLVED	RESONANCE INTEGRAL CALCULATION BW 2000			305	

A	7044 F+MAP RS P T= GAROL, RESONANCE OVERLAP AND LATTICE EFFECTS	G	219
RS P =	PSEUDO, STATISTICAL RESONANCE PARAMETER CALC	GGA 1108 F4	292
P =	EXT, X-SECTIONS FROM B-W RESONANCE PARAMETERS	WAL 7094 F2 RS	238
F2 RS P =	DOPIE, RESOLVED RESONANCE X-SECTION CALC	GESJ 2000	177
E, X-SECTION CALC ELASTIC SCAT	RESONANCES AI 7090 F+MAP RS PLT=	AILMO	147
E, X-SECTION CALC ELASTIC SCAT	RESONANCES AI 360 F4 RS PLT=	AILMO	147
CELL BAPL 6600 F4 RS P T=	RESQ2/RESQ0/DBF1, RESONANCE INTEGRAL HEX		R285
7044 F+MAP RSBP =	UNPACK, RETRIEVAL FROM SCISRS X-SECTION TAPE	GA	206
ING GA 7044 F4 RS P T=	REVISED-GAD, FUEL CYCLE ANALYSIS W/REFUEL		223
CELL ANL 3600 F63 RSBP T=	RIFF-RAFF, RESONANCE INTEGRAL CALC 2-REG		213
RS =	SCAR1, SCATTER FROM A RING SNAP GEOMETRY	AI 7090 F2	111
DIFFUSION SLAB CYLINDER SPHERE	ROA 360 F4 RS PLT=	FIRE2, 1-D AGE-D	9
DEPENDENT KINETICS W/FEEDBACK	ROA 360 F4 RS P =	AIREK3, SPACE-IN	121
M PP 7040 F+MAP RS P T=	RSAC, RADIOLOGICAL SAFETY ANALYSIS PROGRA		265
, 2-D MULTI-GP DIFFUSION XY RZ	RTH WAL 7094 F4 RS P T=	VARI-QUIR3	264
PERT4, 2-D PERTURBATION XY RZ	RTHETA GEOMETRY BNWL 1108 F4 RS P =		304
HTGR CONTAINMENT PRESSURE POST	RUPTURE GA 7044 F4 RS T=	PRECON,	228
ARS, 2-D EXCURSION CALCULATION	R-Z GEOMETRY APDA 7094 F4 RS P =	M	293
2-D MULTI-GP SN APPROXIMATION	RZ PW 1604 F63 RS P T=	FORTAN TDC,	161
N, 2-D FEW-GP S4 APPROXIMATION	RZ GEOMETRY LRLL 709 F2 RS =	FIR	7
B, 2-D FEW-GP DIFFUSION BURNUP	RZ GEOMETRY GA 7090 F+MAP RSBP T=	DD	99
LE4, 2-D MULTI-GP DIFFUSION XY	RZ GEOMETRY GA 7044 F+MAP RSBP T=	GAMB	222
LE5, 2-D MULTI-GP DIFFUSION XY	RZ GEOMETRY GA 1108 F+BAL RS P =	GAMB	222
IR3, 2-D MULTI-GP DIFFUSION XY	RZ RTH WAL 7094 F4 RS P T=	VARI-QU	264
= PERT4, 2-D PERTURBATION XY	RZ RTHETA GEOMETRY BNWL 1108 F4 RS P		304
AMTEC2, MULTI-GP CONSTANT CALC	O TO 10 MEV BNWL 7090 F+MAP RS PLT=	G	185
AMTEC2, MULTI-GP CONSTANT CALC	O TO 10 MEV BNWL 1107 F4 RS PLT=	G	185
2 RS =	FIRN, 2-D FEW-GP S4 APPROXIMATION RZ GEOMETRY LRLL 709 F		7
= S4 CYL CELL CODE, 1-D 1-GP	S4 APPROXIMATION AI 7090 F2 RS		53
= S4 CYL CELL CODE, 1-D 1-GP	S4 APPROXIMATION CDC 1604 F63 RS		53
ON AI 7090 F2 RS =	S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATI		53
ON CDC 1604 F63 RS =	S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATI		53
AI 7090 F2 RS P =	BAM, S4 CYL CELL CODE AND TEMPEST COMBINATION		108
= GRDWRK, GRID GENERATION FOR	SAFE PROGRAMS GGA 1108 F4 RS P		296
TUDY GGA 1108 F4 RS P T=	SAFE-3D, 3-D COMPOSITE STRUCTURE STRESS S		332
LOAD GA 7044 F4 RS P T=	SAFE-AXISYM, STRESS ANALYSIS AXISYMMETRIC		251
E GGA 1108 F4 RS P =	SAFE-CREEP, VISCOELASTIC ANALYSIS CONCRET		300
QAD GA 7044 F4 RS P T=	SAFE-PCRS, STRESS ANALYSIS AXISYMMETRIC L		250
DIES GA 1108 F4 RS P =	SAFE-PLANE, PLANE STRESS ANALYSIS, 2-D BO		252
GA 1108 F4 RS P =	SAFE-SHELL, STRESS ANALYSIS THIN SHELLS		253
AP RS P T=	RSAC, RADIOLOGICAL SAFETY ANALYSIS PROGRAM PP 7040 F+M		265
ENTRY AI 7090 F2 RS =	SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOM		52
FT KE 7094 F+MAP RSBP T=	SATURATED BLOWDOWN1, BLOWDOWN ANALYSIS LO		200
7044 F4 RSBP T=	CROSSPLOT, SC4020 PLOTS FROM X-SECTION TAPES GA		207
WAL 7094 F2 RS T=	CPS, SC4020 PLOTS FROM SCISRS X-SECTION TAPES		239
AI 7090 F2 RS =	SCAR1, SCATTER FROM A RING SNAP GEOMETRY		111
EOM AI 7090 F2 RS =	SCARF2, SCATTER FROM RADIATOR FINS SNAP G		110
7090 F2 RS =	SCARF2, SCATTER FROM RADIATOR FINS SNAP GEOM AI		110
I 7090 F2 RS =	SCAR1, SCATTER FROM A RING SNAP GEOMETRY A		111
RS T=	SUMMIT, CRYSTALLINE SCATTERING KERNEL CALC GA 7090 F2		56
7044 F4 RS =	FLANGE1, SCATTERING LAW X-SECTION CALCULATION GA		247
F4 RS P =	GASKET, THERMAL SCATTERING LAW CALCULATION GA 1108		263
SBP =	UNPACK, RETRIEVAL FROM SCISRS X-SECTION TAPE GA 7044 F+MAP R		206
S T=	CPS, SC4020 PLOTS FROM SCISRS X-SECTION TAPES WAL 7094 F2 R		239
LOAD BAPL 6600 F4 RS P T=	SEALSHL2, SHELL STRESS ANALYSIS AXISYM	R282	
-D FEW-GP DIFFUSION POWER DIST	SEARCH GA 7044 F4 RS P T=	GASP2, 1	220
1-D BURNUP POWER DISTRIBUTION	SEARCH GGA 1108 F4 RS P T=	GASP7,	319
F+MAP RS P T=	DFSR, DATA FILE SERVICE ROUTINES ENDF TAPES BNL 7090		236
600 F4 RS P T=	SEALSHL2, SHELL STRESS ANALYSIS AXISYM LOAD BAPL 6		R282
RS P T=	SOR2, STRESS ANALYSIS SHELLS OF REVOLUTION KAPL 6600 F4		R 80
FE-SHELL, STRESS ANALYSIS THIN SHELLS	GA 1108 F4 RS P =	SA	253

FARSE1A, DOSE RATE FROM SNAP	SHIELD LEAKAGE	AI	7090 F2	RS	=	91
BNWL 7090 F2	RSBPLT= MAC,	SHIELD DESIGN MULTIGROUP SLAB GEOMETRY				143
ANL 3600 F63	RSBPLT= MAC,	SHIELD DESIGN MULTIGROUP SLAB GEOMETRY				143
AI 7094 F2	RS P = SHOE,	SHIELD WEIGHT OPTIMIZATION DOSE CALC				197
RS P T= MUSCAT, VIEW FACTOR		SHIELDING CODE CAVITY GEOM GA	1108 F4			259
ION AI 7090 F2	RS =	SHOCK, SPACE POWER PLANT DESIGN OPTIMIZAT				114
C AI 7094 F2	RS P =	SHOE, SHIELD WEIGHT OPTIMIZATION DOSE CAL				197
360 F+BAL RS P T= AIROS2A,		SIMULATION OF REACTOR DYNAMICS	AI			326
P RS P T= AISITE2, PARAMETRIC		SITE REQUIREMENT STUDY	AI	7094 F+FA		172
RS P T= AISITE2, PARAMETRIC		SITE REQUIREMENT STUDY	AI	360 F4		172
63 RS P T= R153, PARAMETRIC		SITE REQUIREMENT STUDY	ANL	3600 F		172
N AI 7090 F2	RS LT=	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETIO				58
N CDC 1604 F63	RS LT=	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETIO				58
N AI 360 F4	RSBP T=	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETIO				58
A, 2-D 2-GP DIFFUSION CYLINDER		SLAB ORNL 7090 F2	RS P T= EQUIPOISE3			87
A, 2-D 2-GP DIFFUSION CYLINDER		SLAB CDC 1604 F63	RS P T= EQUIPOISE3			87
PLT= FIRE2, 1-D AGE-DIFFUSION		SLAB CYLINDER SPHERE ROA	360 F4	RS		9
T= FOG, 1-C FEW-GP DIFFUSION		SLAB CYLINDER SPHERE AI	7090 F2	RS P		28
T= FOG, 1-D FEW-GP DIFFUSION		SLAB CYLINDER SPHERE CDC	1604 F63	RS P		28
AIM6, 1-D MULTI-GP DIFFUSION		SLAB CYL SPHERE AI	7090 F+FA	RS PLT=		29
AIM6, 1-D MULTI-GP DIFFUSION		SLAB CYL SPHERE CDC	1604 F63	RS PLT=		29
IPOISE3, 2-D 2-GROUP DIFFUSION		SLAB CYL ORNL 7090 F2	RS P T= EQU			39
GRAND, 2-D FEW-GROUP DIFFUSION		SLAB CYLINDER ORNL 7090 F2	RS P T= 20			40
GRAND, 2-D FEW-GROUP DIFFUSION		SLAB CYLINDER CDC	1604 F63	RS P T= 20		40
ULCER, 1-D MULTI-GP DIFFUSION		SLAB CYL SPHERE AI	7090 F+FA	RS LT=		118
FAIM, 1-D MULTI-GP DIFFUSION		SLAB CYL SPHERE AI	7090 F+FA	RS LT=		120
FAIM, 1-D MULTI-GP DIFFUSION		SLAB CYL SPHERE CDC	1604 F63	RS PLT=		120
FAIMCS, 1-D MULTI-GP DIFFUSION		SLAB CYL SPHERE BOEC	360 F4	RS P T=		120
W-DSN, 1-D MULTI-GP SN APPROX		SLAB CYL SPHERE UK-W	7090 F2	RS P T=		132
MGDSN, 1-D MULTI-GP SN APPROX		SLAB CYL SPHERE PW	1604 LAG1	RS P =		211
= HFN, 1-D MULTI-GP DIFFUSION		SLAB CYL SPHERE BNWL	1107 F4	RS P T		241
MACH1, 1-D MULTI-GP DIFFUSION		SLAB CYL SPHERE ANL	3600 F63	RS P T=		262
GRACE1, GAMMA-RAY ATTENUATION		SLAB GEOMETRY AI	7090 F2	RS P =		45
GRACE1, GAMMA-RAY ATTENUATION		SLAB GEOMETRY CDC	1604 F63	RS P =		45
AIL, 1-D 1-GP SN APPROXIMATION		SLAB GEOMETRY AI	7090 F2	RS = S		52
1-D FEW-GP SN DOUBLE SN APPROX		SLAB GEOM PP	7090 F2	RS P = MIST,		59
MAC, SHIELD DESIGN MULTIGROUP		SLAB GEOMETRY BNWL	7090 F2	RSBPLT=		143
MAC, SHIELD DESIGN MULTIGROUP		SLAB GEOMETRY ANL	3600 F63	RSBPLT=		143
P RS PLT= TYCHE3, MONTE CARLO		SLOWING-DOWN DENSITY CALC	AI	7094 F+FA		149
CO RSBP = 2DXY, 2-D MULTI-GP		SN APPROXIMATION XY GEOM	AGN	7090 FLO		18
F2 RS = SAIL, 1-D 1-GP		SN APPROXIMATION SLAB GEOMETRY	AI	7090		52
= MIST, 1-D FEW-GP SN DOUBLE		SN APPROX SLAB GEOM PP	7090 F2	RS P		59
-HAPC-S13, 1-D MULTI-GP DOUBLE		SN APPROX BNWL	7090 FLO	CO RSBP T= GE		75
RS P T= W-DSN, 1-D MULTI-GP		SN APPROX SLAB CYL SPHERE	UK-W	7090 F2		132
RS P T= TOPIC, 1-D FEW-GP		SN APPROXIMATION CYLINDER	PP	7040 F4		148
RS P T= TOPIC, 1-D FEW-GP		SN APPROXIMATION CYLINDER	PP	7090 F2		148
T= FORTRAN TDC, 2-D MULTI-GP		SN APPROXIMATION RZ	PW	1604 F63	RS P	161
RS P = MGDSN, 1-D MULTI-GP		SN APPROX SLAB CYL SPHERE	PW	1604 LAG1		211
RS P = SNC, CALCULATION OF		SN CONSTANTS FOR DSN TDC	PW	1604 F63		189
2 RS P = MIST, 1-D FEW-GP		SN DOUBLE SN APPROX SLAB GEOM PP	7090 F			59
W 1604 F63 RS = CSP2A,		SN X-SECTION LIBRARY TAPE PREPARATION	P			193
PW 1604 F63 RS = CSP1,		SN X-SECTION LIBRARY TAPE PREPARATION				194
F2, SCATTER FROM RADIATOR FINS		SNAP GEOM AI	7090 F2	RS = SCAR		110
= SCAR1, SCATTER FROM A RING		SNAP GEOMETRY AI	7090 F2	RS		111
SNAPKIN5/5A, 1-REGION KINETICS		SNAP GEOMETRY AI	7090 F2	RS =		122
RATRAP, DOSE RATE CALCULATION		SNAP GEOMETRY AI	7090 F2	RS P =		141
ORTIMER, DOSE RATE CALCULATION		SNAP GEOMETRY AI	7090 F2	RS = M		142
= FARSE1A, DOSE RATE FROM		SNAP SHIELD LEAKAGE	AI	7090 F2	RS	91
TRY AI 7090 F2	RS =	SNAPKIN5/5A, 1-REGION KINETICS	SNAP GEOME			122
CALC ANL 3600 F36	RS P T=	SNARG-1D, 1-D MULTI-GP DISCRETE ORDINATE				288
TDC PW 1604 F63	RS P =	SNC, CALCULATION OF SN CONSTANTS FOR DSN				189

BURP4, GAMMA-RAY PHOTOFRACTION	SOLID CRYSTAL	UM	7090 MAD	RSB	=	169
RS P = AXTHRM, HEAT TRANSFER	SOLID FUEL TUBE BUNDLES	LASL	7094 F2			183
NTS LRLL 7090 F+FAP RS P T=	SOPHIST1/2/5, MULTI-GP TRANSFER COEFFICIE					160
N KAPL 6600 F4 RS P T=	SOR2, STRESS ANALYSIS SHELLS OF REVOLUTIO					R 80
RP1, DETECTOR EFFICIENCY POINT	SOURCE	UM	7090 MAD	RSB	= BU	164
URP2, DETECTOR EFFICIENCY DISK	SOURCE	UM	7090 MAD	RSB	= 8	165
RP3, DETECTOR EFFICIENCY POINT	SOURCE	UM	7090 MAD	RSB	= BU	166
PN, ANALYSIS OF PULSED NEUTRON	SOURCE DATA	GESJ	635 F4	RS P T=	EX	258
RS = DANCOFF JR, MODERATOR	SPACE CHORD DIST FUNCT	AEG	7090 F2			150
I 7090 F2 RS = CROCK,	SPACE POWER PLANT DESIGN OPTIMIZATION	A				112
I 7090 F2 RS = SHOCK,	SPACE POWER PLANT DESIGN OPTIMIZATION	A				114
ANL 3600 F63 RSBP = R102,	SPACE-INDEPENDENT INVERSE KINETICS CALC					168
WAL 7094 F4 RS = R102,	SPACE-INDEPENDENT INVERSE KINETICS CALC					168
7090 F2 RS = AIREK3,	SPACE-INDEPENDENT KINETICS W/FEEDBACK	AI				121
A 360 F4 RS P = AIREK3,	SPACE-INDEPENDENT KINETICS W/FEEDBACK	RO				121
I 7094 F+MAP RS P T= AIROS,	SPACE-INDEPENDENT KINETICS W/FEEDBACK	A				163
ANL 3600 F63 RSBP = R101,	SPACE-INDEPENDENT KINETICS KEX OPTIONS					255
WAL 7094 F4 RS P = R101,	SPACE-INDEPENDENT KINETICS KEX OPTIONS					255
1108 F4 RS P = GAPOTKIN,	SPACE-INDEPENDENT REACTOR KINETICS	GGA				317
F4 RS P T= WIGL2, 1-D 2-GP	SPACE-TIME DIFFUSION 3-GEOM	BAPL	6600			R274
S GESJ 2000 F2 RS P =	SPARTA, SPATIALLY-AVERAGED DOPPLER EFFECT					178
SJ 2000 F2 RS P = SPARTA,	SPATIALLY-AVERAGED DOPPLER EFFECTS	GE				178
AI 7094 F4 RS P T= TSN,	SPATIALLY-DEPENDENT REACTOR KINETICS					309
AI 360 F4 RS P T= TSN,	SPATIALLY-DEPENDENT REACTOR KINETICS					309
1108 F4 RS P T= GAF/GAR,	SPECTRA AND GROUP-AVERAGED X-SEC CALC	GGA				316
D 7090 F+FAP RSBPLT= LEOPARD,	SPECTRA CALCULATION WITH DEPLETION	WAP				279
LTI-GP X-SECTIONS FAST THERMAL	SPECTRA GGA 1108 F4 RS PLT=	GGC4, MU				298
OCESSING OF ANALYZER GAMMA-RAY	SPECTRA GGA 1108 F4 RS P T=	TOAD, PR				333
094 F+FAP RS P T= EPITHERMOS,	SPECTRUM AND X-SECTION CALCULATION	GEV	7			201
APD 7094 F+MAP RS PLT= LASER,	SPECTRUM CALC WITH BURNUP CYL LATTICE	W				249
O F2 RSBPLT= AGN-GAM, FAST	SPECTRUM MULTI-GP CONSTANT CALC	AGN	709			204
RS PLT= GAM1, FAST NEUTRON	SPECTRUM X-SECTION CALC	GA	7090 F2			33
RS PLT= GAM1, FAST NEUTRON	SPECTRUM X-SECTION CALC	CDC	1604 F63			33
RS LT= GAM1, FAST NEUTRON	SPECTRUM X-SECTION CALC	ANL	3600 F63			33
RS PLT= FORM, FAST NEUTRON	SPECTRUM X-SECTION CALC	AI	7090 F2			51
RS PLT= FORM, FAST NEUTRON	SPECTRUM X-SECTION CALC	CDC	1604 F63			51
RS P T= THERMOS(ANL), THERMAL	SPECTRUM X-SECTION CALC	ANL	3600 F63			184
LT= TEMPEST2, THERMAL NEUTRON	SPECTRUM X-SECTIONS	AI	7090 F+FAP RS P			50
LT= TEMPEST2, THERMAL NEUTRON	SPECTRUM X-SECTIONS	CDC	1604 F63 RS P			50
LT= TEMPEST2, THERMAL NEUTRON	SPECTRUM X-SECTIONS	BOEC	360 F4 RS P			50
LT= QUICKIE, INFINITE MEDIUM	SPECTRUM X-SECTIONS	AI	7090 F+FAP RS			119
UPLED NEUTRONICS-HYDRODYNAMICS	SPH PW 1604 F63 RS P =	AX-TNT, CO				191
-D AGE-DIFFUSION SLAB CYLINDER	SPHERE ROA 360 F4 RS PLT=	FIRE2, 1				9
FEW-GP DIFFUSION SLAB CYLINDER	SPHERE AI 7090 F2 RS P T=	FOG, 1-D				28
FEW-GP DIFFUSION SLAB CYLINDER	SPHERE CDC 1604 F63 RS P T=	FOG, 1-D				28
-D MULTI-GP DIFFUSION SLAB CYL	SPHERE AI 7090 F+FAP RS PLT=	AIM6, 1				29
-D MULTI-GP DIFFUSION SLAB CYL	SPHERE CDC 1604 F63 RS PLT=	AIM6, 1				29
CE2, GAMMA-RAY ATTENUATION CYL	SPHERE GEOM AI 7090 F2 RS P =	GRA				46
CE2, GAMMA-RAY ATTENUATION CYL	SPHERE GEOM CDC 1604 F63 RS P =	GRA				46
UPLED NEUTRONICS-HYDRODYNAMICS	SPHERE CDC 3600 F63 RSBP =	AX1, CO				102
-D MULTI-GP DIFFUSION SLAB CYL	SPHERE AI 7090 F+FAP RS LT=	ULCER, 1				118
-D MULTI-GP DIFFUSION SLAB CYL	SPHERE AI 7090 F+FAP RS LT=	FAIM, 1				120
-D MULTI-GP DIFFUSION SLAB CYL	SPHERE CDC 1604 F63 RS PLT=	FAIM, 1				120
-D MULTI-GP DIFFUSION SLAB CYL	SPHERE BOEC 360 F4 RS P T=	FAIMOS, 1				120
UPLED NEUTRONICS-HYDRODYNAMICS	SPHERE LRLL 7090 F2 RS P =	CONEC, CO				129
-D MULTI-GP SN APPROX SLAB CYL	SPHERE UK-W 7090 F2 RS P T=	W-DSN, 1				132
-D MULTI-GP SN APPROX SLAB CYL	SPHERE PW 1604 LAG1 RS P =	MGDSN, 1				211
-D MULTI-GP DIFFUSION SLAB CYL	SPHERE BNWL 1107 F4 RS P T=	HFN, 1				241
-D MULTI-GP DIFFUSION SLAB CYL	SPHERE ANL 3600 F63 RS P T=	MACH1, 1				262
DOSE RATE FROM A CLOUD	SRL 360 F4 RS P =	CLOUD, GAMMA-RAY				47
GGA 1108 F4 RS P = GASA,	STABILITY ANALYSIS REACTOR KINETICS EQNS					290

A 1108 F4	RS P = PSEUDO,	STATISTICAL RESONANCE PARAMETER CALC	GG	292
I 7090 F2	RS = FUGUE,	STEADY-STATE TEMPERATURE VOID FRACTION	A	48
	RS P T= HEATING2, TRANSIENT	STEADY-STATE HEAT TRANSFER	AI 7094 F2	198
ESJ 635 F4	RS P T= MANTA,	STEADY-STATE THERMAL-HYDRAULIC ANALYSIS	G	256
7044 F4	RS P T= STMGEN,	STEAM GENERATOR DESIGN CRITERIA COSTS	GA	227
APL 2000 F4	RS = WATER,	STEAM TABLES 14.5-14,500PSIA 32-472DEGF	B	R267
6600 F4	RS P T= M0899/HOH,	STEAM TABLES 14.5-2538 PSIA	BAPL	R294
OSTS GA 7044 F4	RS P T=	STMGEN, STEAM GENERATOR DESIGN CRITERIA	C	227
KAPL 6600 F4	RS P T= SOR2,	STRESS ANALYSIS SHELLS OF REVOLUTION		R 80
F+FAP RS	T= 4RESTRAINT PIPE	STRESS, MAXIMUM MOMENT CALC	AI 7090	109
RS P T= ELBOW, FUEL ELEMENT		STRESS ANALYSIS STUDY	GA 7044 F4	229
44 F4	RS P T= PIZZA, PLANE	STRESS ANALYSIS OF 2-D BODIES	GA 70	230
7044 F4	RS P T= SAFE-PCRS,	STRESS ANALYSIS AXISYMMETRIC LOAD	GA	250
44 F4	RS P T= SAFE-AXISYM,	STRESS ANALYSIS AXISYMMETRIC LOAD	GA 70	251
RS P = SAFE-PLANE, PLANE		STRESS ANALYSIS, 2-D BODIES	GA 1108 F4	252
108 F4	RS P = SAFE-SHELL,	STRESS ANALYSIS THIN SHELLS	GA 1	253
PL 6600 F4	RS T= CYGR02,	STRESS ANALYSIS CYL FUEL ELEMENT	BA	R266
RS P T= SEALSHLL2, SHELL		STRESS ANALYSIS AXISYM LOAD	BAPL 6600 F4	R282
RS P T= M0457(PIPE), ELASTIC		STRESS OF PIPING SYSTEM	BAPL 6600 F4	R329
FE-3D, 3-D COMPOSITE STRUCTURE		STRESS STUDY	GGA 1108 F4	RS P T= SA
ATION BW 2000 F4	RS =	STRIP, RESOLVED RESONANCE INTEGRAL CALCUL		305
C GA 7090 F2	RS T=	SUMMIT, CRYSTALLINE SCATTERING KERNEL CAL		56
4 RS P T= FIGRO, LSBR FUEL		SWELLING TEMPERATURE STUDY	BAPL 6600 F	R272
CORPORATION, TRANSFER FUNCTION		SYNTHESIS ANL 3600 F63	RSBP T= TRAFI	135
T= 2DXYL, 3-D MULTI-GP FLUX		SYNTHESIS PROGRAM CYL PW 1604 F63	RS	192
BISYN, 2-D MULTI-GP DIFFUSION		SYNTHESIS CALC GESJ 635 F4	RS PLT=	287
00 F4	RS = WATER, STEAM	TABLES 14.5-14,500PSIA 32-472DEGF	BAPL 20	R267
4 RS P T= M0899/HOH, STEAM		TABLES 14.5-2538 PSIA	BAPL 6600 F	R294
ETRIEVAL FROM SCISRS X-SECTION		TAPE GA 7044 F+MAP	RSBP = UNPACK, R	206
RS = WED, W-DSN OUTPUT		TAPE EDIT REACTION RATES	UK-W 7090 F2	133
= CSP2A, SN X-SECTION LIBRARY		TAPE PREPARATION PW 1604 F63	RS	193
= CSP1, SN X-SECTION LIBRARY		TAPE PREPARATION PW 1604 F63	RS	194
T, SC4020 PLOTS FROM X-SECTION		TAPES GA 7044 F4	RSBP T= CROSSPLO	207
ATA FILE SERVICE ROUTINES ENDF		TAPES BNL 7090 F+FAP	RS P T= DFSR, D	236
20 PLOTS FROM SCISRS X-SECTION		TAPES WAL 7094 F2	RS T= CPS, SC40	239
LATION OF SN CONSTANTS FOR DSN		TDC PW 1604 F63	RS P = SNC, CALCU	189
RS = TDP, 2-D PERTURBATION		TDC OR 2DXY FLUX INPUT	PW 1604 F63	199
1604 F63	RS P T= FORTRAN	TDC, 2-D MULTI-GP SN APPROXIMATION	RZ PW	161
PUT PW 1604 F63	RS =	TDP, 2-D PERTURBATION TDC OR 2DXY FLUX IN		199
OGRAM LRC 7090 F+MAP	RS P T=	TDSN, 2-D MULTIGROUP DISCRETE ORDINATE	PR	312
ON GGA 1108 F4	RS P T=	TEMCO7, TEMPERATURE COEFFICIENT CALCULATI		320
C GA 7044 F4	RS P T=	TEMCO, 1-D FEW-GP DIFFUSION TEMP COEF CAL		225
= TEMCO, 1-D FEW-GP DIFFUSION		TEMP COEF CALC	GA 7044 F4	RS P T
RS = FUGUE, STEADY-STATE		TEMPERATURE VOID FRACTION	AI 7090 F2	48
63	RSBP T= ARGUS, TRANSIENT	TEMPERATURE CALC CYLINDER	ANL 3600 F	152
BP T= PTH1, BLOWDOWN PRESSURE		TEMPERATURE HISTORY	KE 7094 F2	RS
CORE, CORE CONFIGURATION FUEL		TEMPERATURE CODE	GA 7044 F4	RS =
T= FIGRO, LSBR FUEL SWELLING		TEMPERATURE STUDY	BAPL 6600 F4	RS P
6600 F4	RS P T= LION, 3-D	TEMPERATURE DISTRIBUTION PROGRAM	KAPL	R299
A 1108 F4	RS P T= TEMCO7,	TEMPERATURE COEFFICIENT CALCULATION	GG	320
IGNS AI 7090 F+FAP	RS PLT=	TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECT		50
IONS CDC 1604 F63	RS PLT=	TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECT		50
IONS BOEC 360 F4	RS PLT=	TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECT		50
= BAM, S4 CYL CELL CODE AND		TEMPEST COMBINATION	AI 7090 F2	RS P
7090 F+FAP	RS PLT= TEMPEST2,	THERMAL NEUTRON SPECTRUM X-SECTIONS	AI	50
1604 F63	RS PLT= TEMPEST2,	THERMAL NEUTRON SPECTRUM X-SECTIONS	CDC	50
360 F4	RS PLT= TEMPEST2,	THERMAL NEUTRON SPECTRUM X-SECTIONS	BOEC	50
0 F63	RS P T= THERMOS(ANL),	THERMAL SPECTRUM X-SECTION CALC	ANL 360	184
= GAZELLE5, GAS-COOLED CORE		THERMAL DESIGN STUDY	GA 7044 F4	RS P
RS P T= MANTA, STEADY-STATE		THERMAL-HYDRAULIC ANALYSIS	GESJ 635 F4	256
1108 F4	RS P = GASKET,	THERMAL SCATTERING LAW CALCULATION	GA	263

GGC4, MULTI-GP X-SECTIONS FAST	THERMAL SPECTRA GGA 1108 F4	RS PLT=	298
= NURLOC-1.0, LOSS-OF-COOLANT	THERMAL ANALYSIS BMI 6400 F4	RS P T	328
CALC ANL 3600 F63 RS P T=	THERMOS(ANL), THERMAL SPECTRUM X-SECTION		184
7094 F2 RS PLT= CYCLOPS1,	THERMODYNAMIC CYCLE ANALYSIS AI		244
0 F4 RSB LT= CLIP, FORM OR	THREDES LIBRARY UTILITY ROUTINE AI	36	271
EM AI 360 F4 RSBP T=	THREDES, 1-D FEW-GP DIFFUSION DESIGN SYST		273
7090 F+FAP RS P T= VARI-QUIR,	TIME-DEP 2-D MULTI-GP DIFFUSION WAL		212
RS P T= GAKIN, 1-D MULTIGROUP	TIME-DEPENDENT DIFFUSION GGA 1108 F4		310
ECTRA GGA 1108 F4 RS P T=	TOAD, PROCESSING OF ANALYZER GAMMA-RAY SP		333
ER PP 7040 F4 RS P T=	TOPIC, 1-D FEW-GP SN APPROXIMATION CYLIND		148
ER PP 7090 F2 RS P T=	TOPIC, 1-D FEW-GP SN APPROXIMATION CYLIND		148
ESIS ANL 3600 F63 RSBP T=	TRAFICORPORATION, TRANSFER FUNCTION SYNTH		135
3 RSBP T= TRAFICORPORATION,	TRANSFER FUNCTION SYNTHESIS ANL 3600 F6		135
S P T= SOPHIST1/2/5, MULTI-GP	TRANSFER COEFFICIENTS LRLL 7090 F+FAP R		160
1604 F63 RS P = CMPXMAT,	TRANSFER FUNCTION EVALUATION PW		188
= AGN-SIGMA, CALC OF MULTI-GP	TRANSFER MATRICES AGN 7090 F2 RS P T		243
NL 3600 F63 RSBP T= ARGUS,	TRANSIENT TEMPERATURE CALC CYLINDER A		152
7094 F2 RS P T= HEATING2,	TRANSIENT STEADY-STATE HEAT TRANSFER AI		198
4 F2 RS P T= TRANS-FUGUE1,	TRANSIENT FLOW AND HEAT TRANSFER AI 709		268
6600 F4 RS P T= HOT2, 2-D	TRANSIENT HEAT CONDUCTION PROGRAM BAPL	R286	
S P T= FLOT1(MO219), PWR FLOW	TRANSIENT ANALYSIS BAPL 6600 F4 R	R331	
NSFER AI 7094 F2 RS P T=	TRANS-FUGUE1, TRANSIENT FLOW AND HEAT TRA		268
AI 7094 F+MAP RS PLT=	TRIX1, RESONANCE INTEGRAL X-SECTION CALC		208
AI 7094 F4 RS P T=	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS		309
AI 360 F4 RS P T=	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS		309
FLU, HEAT TRANSFER MOLTEN FUEL	TUBE BUNDLES LASL 7094 F2 RS P = AX		182
THRM, HEAT TRANSFER SOLID FUEL	TUBE BUNDLES LASL 7094 F2 RS P = AX		183
CALC GA 7090 F+FAP RSB T=	TUZ, UNRESOLVED REGION RESONANCE INTEGRAL		42
CALC AI 7094 F+FAP RS PLT=	TYCHE3, MONTE CARLO SLOWING-DOWN DENSITY		149
QUID BLOWDOWN ANALYSIS LOFT	UG 360 F4 RS P = WATER-HAMMER, LI		278
COMPUTER-PRODUCED WIRING LISTS	UHTREX LASL 7090 F2 RS P = WIREX,		315
D MULTI-GP DIFFUSION PROGRAM	UK-R 7090 F+FAP RSBPLT= CRAM, 1-D AND 2-		103
-GP SN APPROX SLAB CYL SPHERE	UK-W 7090 F2 RS P T= W-DSN, 1-D MULTI		132
T TAPE EDIT REACTION RATES	UK-W 7090 F2 RS = WED, W-DSN OUTPU		133
HERE AI 7090 F+FAP RS LT=	ULCER, 1-D MULTI-GP DIFFUSION SLAB CYL SP		118
EFFICIENCY POINT SOURCE	UM 7090 MAD RSB = BURP1, DETECTOR		164
EFFICIENCY DISK SOURCE	UM 7090 MAD RSB = BURP2, DETECTOR		165
EFFICIENCY POINT SOURCE	UM 7090 MAD RSB = BURP3, DETECTOR		166
PHOTOFRACTION SOLID CRYSTAL	UM 7090 MAD RSB = BURP4, GAMMA-RAY		169
PHOTOFRACTION WELL CRYSTAL	UM 7090 MAD RSB = BURP5, GAMMA-RAY		170
OUN DISCRETE ORDNATE CALC	UNC 1604 F63 RS PLT= DTF, 1-D MULTIGR		144
TI-GP DISCRETE ORDNATE CODE	UNC-LASL 1604 F63 RS = 2DF, 2-D MUL		173
APE GA 7044 F+MAP RSBP =	UNPACK, RETRIEVAL FROM SCISRS X-SECTION T		206
GA 7090 F+FAP RSB T= TUZ,	UNRESOLVED REGION RESONANCE INTEGRAL CALC		42
7090 F2 RS P = AIMFIRE,	URANIUM FUEL CYCLE COST ANALYSIS AI		55
LT= MOMUS, X-SECTION LIBRARY	UTILITY PROGRAM AI 7094 F+FAP RS		159
T= XLBIT, X-SECTION LIBRARY	UTILITY ROUTINE ANL 3600 F63 RSBP		181
CLIP, FORM OR THREDES LIBRARY	UTILITY ROUTINE AI 360 F4 RSB LT=		271
N WAL 7090 F+FAP RS P T=	VARI-QUIR, TIME-DEP 2-D MULTI-GP DIFFUSIO		212
RTH WAL 7094 F4 RS P T=	VARI-QUIR3, 2-D MULTI-GP DIFFUSION XY RZ		264
1108 F4 RS P T= MUSCAT,	VIEW FACTOR SHIELDING CODE CAVITY GEOM GA		259
108 F4 RS P = SAFE-CREEP,	VISCOELASTIC ANALYSIS CONCRETE GGA 1		300
UGUE, STEADY-STATE TEMPERATURE	VOID FRACTION AI 7090 F2 RS = F		48
F4 RS P = AVOID, ANNULAR	VOID X-SECTION CALCULATION GEC 625		276
PENDENT INVERSE KINETICS CALC	WAL 7094 F4 RS = R102, SPACE-INDE		168
DEP 2-D MULTI-GP DIFFUSION	WAL 7090 F+FAP RS P T= VARI-QUIR, TIME-		212
FROM B-W RESONANCE PARAMETERS	WAL 7094 F2 RS P = EXT, X-SECTIONS		238
S FROM SCISRS X-SECTION TAPES	WAL 7094 F2 RS T= CPS, SC4020 PLOT		239
PENDENT KINETICS KEX OPTIONS	WAL 7094 F4 RS P = R101, SPACE-INDE		255
MULTI-GP DIFFUSION XY RZ RTH	WAL 7094 F4 RS P T= VARI-QUIR3, 2-D		264
Y GA 7044 F4 RS P =	WAMPUM, FUEL CYCLE COSTS PERFORMANCE STUD		224

CALC WITH BURNUP CYL LATTICE	WAPD 7094 F+MAP RS PLT=	LASER, SPECTRUM	249
CALCULATION WITH DEPLETION	WAPD 7090 F+FAP RSBPLT=	LEOPARD, SPECTRA	279
FT UG 360 F4 RS P =	WATER-HAMMER, LIQUID BLOWDOWN ANALYSIS LO		278
2DEGF BAPL 2000 F4 RS =	WATER, STEAM TABLES 14.5-14,500PSIA 32-47		R267
F63 RS P = RAMES, PARTICLE	WAVE FUNCTION RADIAL INTEGRALS ORNL 1604		335
MICS APDA 7094 F2 RS P =	WEAK EXPLOSION, COUPLED NEUTRON-HYDRODYNA		145
S UK-W 7090 F2 RS =	WED, W-DSN OUTPUT TAPE EDIT REACTION RATE		133
3 RS = WOPEXPRT, REACTOR	WEIGHT OPTIMIZATION STUDY PW 1604 F6		190
94 F2 RS P = SHOE, SHIELD	WEIGHT OPTIMIZATION DOSE CALC AI 70		197
BURP5, GAMMA-RAY PHOTOFRACTION	WELL CRYSTAL UM 7090 MAD RSB =		170
S KE 3600 F4 RSBP =	WHAM, LIQUID-FILLED PIPING SYSTEM ANALYSI		278
ENTRY ORNL 7090 F2 RS P T=	WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOM		32
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