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Supplement 5

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Argonne National Laboratory

ARGONNE CODE CENTER: COMPILATION OF PROGRAM ABSTRACTS

by

M. K. Butler, Pamela Henline,
Marianne Legan, L. Ranzini,
and William J. Snow

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PAGES TO BE INSERTED	PAGES TO BE REMOVED	COMMENT
1-20	1-20	Revised cover, title page, table of contents, preface, and Sections I-IV
445-448	445-448	Reactor Code Abstract 295 revised
517-520	517-520	Reactor Code Abstract 329 revised
579-582	579-582	Reactor Code Abstract 358 revised
599-602	599-602	Reactor Code Abstract 368 revised
689-909	689-813	Reactor Code Abstracts 411 through 449 added; Section VI Library Programs by Classification revised; Section VII KWIC Index revised; References replaced

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217	LEGCCEF3, LEGENDRE COEF CALC FOR ANGULAR DIST	GGA	7044	F4	RS P X	A	287
218	GAVER, ENERGY INTERVAL X-SECTICN AVERAGING	GGA	7044	F4	RS P	L	289
219	GARCL, RESONANCE OVERLAP AND LATTICE EFFECTS	GGA	7044	F+MAP	RS P T	B	291
219	GARCL, RESONANCE OVERLAP AND LATTICE EFFECTS	LER	7094	F4	RS P T	B	291
220	GASP2, 1-D FEW-GP CIFFUSICN POWER DIST SEARCH	GGA	7044	F4	RS P T	C	293
221	RELCAD-FEVER, 1-D FEW-GP DIFFUSICN DEPLETION	GGA	7044	F4	RS P T	D	295
222	GAMBLE4, 2-D MULTI-GP DIFFUSICN XY RZ GECMETRY	GGA	7044	F+MAP	RSBP T	C	297
222	GAMBLE5, 2-D MULTI-GP DIFFUSION XY RZ GECMETRY	GGA	1108	F+BAL	RS P T	C	297
223	REVISEC-GAD, FUEL CYCLE ANALYSIS W/REFUELING	GGA	7044	F4	RS P T	D	299
224	WAMPUM, FUEL CYCLE COSTS PERFORMANCE STUDY	GGA	7044	F4	RS P	D	301
225	TEMCC, 1-D FEW-GP CIFFUSICN TEMP CCEF CALC	GGA	7044	F4	RS P T	C	303
226	OPLS, PCWER PLANT PERFORMANCE AND PRICE STUDY	GGA	7044	F+MAP	RS P T	D	305
227	STMCEN, STEAM GENERATOR DESIGN CRITERIA COSTS	GGA	7044	F4	RS P T	D	307
228	PRECON, HTGR CCNTAINMENT PRESSURE PCST RUPTURE	GGA	7044	F4	RS XT	G	309
229	ELBOW, FUEL ELEMENT STRESS ANALYSIS STUDY	GGA	7044	F4	RS P T	G	311
231	RAD2, HTGR FISSICN PRODUCT ACTIVITY DIST STUDY	GGA	7044	F4	RS P	D	315
232	GAZELLE5, GAS-COOLED CORE THERMAL DESIGN STUDY	GGA	7044	F4	RS P	I	317
233	CORE, CORE CONFIGURATION FUEL TEMPERATURE CODE	GGA	7044	F4	RS	I	319
234	DPC, DATA PREPARATION FCR 2-D CESIGN PROGRAMS	LASL	7090	F2	RS P T	L	321
235	GAMMA-P, PRODUCTICN X-SECTIONS FOR GAMMA-RAYS	GEC	7094	F4	RS P	B	323
236	DFSR, DATA FILE SERVICE ROUTINES ENDF TAPES	BNL	7090	F+FAP	RS P T	M	325
237	BOUNCE, FLUX DIST IN MULTI-PIN FUEL ELEMENT	AGC	7090	F2	RS P	B	327
238	EXT, X-SECTIONS FRCM B-W RESCNANCE PARAMETERS	WANL	7094	F2	RS P	A	329
239	CPS, SC4020 PLCTS FROM SCISRS X-SECTION TAPES	WANL	7094	F2	RSB T	N	331
240	ASSALLT, 2-D MULTI-GP DIFFUSICN DEPLETICN CODE	ORNL	7090	F+FAP	RS P T	D	333
241	HFN, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	BNW	1107	F4	RS P T	C	336
241	RATH, 2- OR 3-C HEAT CONDUCTICN LUMPED MASS	LASL	7094	FAP	RS P T	H	338
242	RATH, 2- OR 3-D HEAT CONDUCTICN LUMPED MASS	LASL	7030	F4	RS T	H	338
243	AGN-SIGMA, CALC OF MULTI-GP TRANSFER MATRICES	AGC	7090	F2	RS P T	B	340
244	CYCLCPS1, THERMODYNAMIC CYCLE ANALYSIS	AI	7094	F2	RS PL T	I	342
246	FLOW-MODEL, MULTI-CHANNEL 2-D 2-PHASE FLOW	AI	7094	F2	RS P	H	344
247	FLANGE1, SCATTERING LAW X-SECTICN CALCULATION	GGA	1108	F5	RS P	A	346
248	COINC, COINCIDENCE COUNTING CATA REDUCTION	ANL	3600	F63	RSBP	O	348
249	LASER, SPECTRUM CALC WITH BURNUP CYL LATTICE	WAPD	7094	F+MAP	RS PLXT	B	350
250	SAFE-PCRS, STRESS ANALYSIS AXISYMMETRIC LOAC	GGA	7044	F4	RS P T	I	353
251	SAFE-AXISYM, STRESS ANALYSIS AXISYMMETRIC LOAD	GGA	7044	F4	RS P T	I	355
252	SAFE-PLANE, PLANE STRESS ANALYSIS, 2-D BODIES	GGA	1108	F4	RS P	I	357
253	SAFE-SHELL, STRESS ANALYSIS THIN SHELLS	GGA	1108	F4	RS P	I	359
254	2PLLS, NON-SPHERICAL OPTICAL MCDEL X-SECTIONS	ANL	3600	F63	RSBP T	A	361
254	2PLLS, NON-SPHERICAL OPTICAL MCDEL X-SECTIONS	AI	7094	F2	RS P T	A	361
255	R1C1, SPACE-INCEPENCENT KINETICS KEX OPTIONS	WANL	7094	F4	RS P	E	363
255	R1C1, SPACE-INCEPENCENT KINETICS KEX OPTIONS	ANL	3600	F63	RSBP	E	363
256	MANTA, STEADY-STATE THERMAL-HYDRAULIC ANALYSIS	NED	635	F4	RS P T	H	365
257	REAX, RESOLVED RESCNANCE EPITHERMAL X-SECTIONS	NED	635	F4	RS P T	B	367
258	EXPN, ANALYSIS OF PULSED NEUTRCN SOURCE CATA	NED	635	F4	RS P T	O	369
259	MUSCAT, VIEW FACTOR SHIELDING CCDE CAVITY GECM	GGA	1108	F4	RS P T	J	371
260	GARGCYLE, FUEL CYCLE ANALYSIS PARTIAL REFUEL	GGA	7044	F4	RS P T	D	373
261	GADCSE,DOSET, HTGR ACCIDENT ANALYSIS DOSE CALC	GGA	1108	F4	RS T	G	375
262	MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	PURD	6500	F4	RS PLXT	C	377
262	MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	ANL	3600	F36	RS P XT	C	377
263	GASKET, THERMAL SCATTERING LAW CALCULATION	GGA	1108	F4	RS P	A	379
264	VARI-QUIR3, 2-C MULTI-GP DIFFUSICN XY RZ RTH	WANL	7094	F4	RS P XT	C	381
265	RSAC, RADIOLOGICAL SAFETY ANALYSIS PROGRAM	PPCO	7040	F+MAP	RS P T	G	383
R266	CYGRO2, STRESS ANALYSIS CYL FUEL ELEMENT	BAPL	6600	F4	RS XT	I	385
R267	WATER, STEAM TABLES 14.5-14,50CPSIA 32-472DEGF	BAPL	6600	F4	RS XT	H	387
268	TRANS-FUGUE1, TRANSIENT FLOW ANC HEAT TRANSFER	AI	7094	F2	RS P T	G	389
269	DTF-BURN, 1-D MULTI-GP DTF4 WITH DEPLETICN	LASL	7030	F4	RS P T	D	391
270	CAESAR4,LIBLST, 1-C MULTI-GP DIFFUSICN + LIB	AI	360	F4	RSBPLXT	C	393
271	CLIP, FORM OR THRECES LIBRARY UTILITY ROUTINE	AI	360	F4	RSB L T	M	396
R272	FIGRO, LSBR FUEL SWELLING TEMPERATURE STUDY	BAPL	6600	F4	RS P XT	H	398
273	THRECES, 1-D FEW-GP DIFFUSICN DESIGN SYSTEM	AI	360	F4	RSBP T	K	400
R274	WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM	GGA	1108	F4	RS P T	F	402
R274	WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM	BAPL	6600	F4	RS P XT	F	402

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R275	PDQ7, 1,2 OR 3-D FEW-GP DIFFUSION DEPLETION	BAPL	6600	F+ASC	RS P XT	D	405
276	AVOID, ANNULAR VOID X-SECTION CALCULATION	GEC	625	F4	RS P	B	407
277	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM	BNL-DP	360	F4	RS PLXT	B	409
277	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM	BNL-DP	7090	F2	RS PLXT	B	409
278	WATER-HAMMER, LIQUID BLOWDOWN ANALYSIS LOFT	UGA	360	F4	RS P	G	411
278	WHAM, LIQUID-FILLED PIPING SYSTEM ANALYSIS	KE	3600	F36	RSBP	G	411
279	LECPARD, SPECTRA CALCULATION WITH DEPLETION	WAPD	360	F4	RS PLXT	B	413
R280	MO8C7, 2-D DIFFUSION ABSORPTION REMOVAL X-SECS	BAPL	6600	F4	RS XT	B	415
281	RABBLE,WLIB,FLAT, RESONANCE ABSORPTION, CELL	ANL	3600	F36	RSBP XT	B	417
R282	SEALSHELL2, SHELL STRESS ANALYSIS AXISYM LOAD	BAPL	6600	F4	RS P XT	I	419
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R285	RESQ2,RESQO,DBF1, RESONANCE INTEGRAL HEX CELL	BAPL	6600	F4	RS P XT	B	425
R286	HOT2, 2-D TRANSIENT HEAT CONDUCTION PROGRAM	BAPL	6600	F4	RS P XT	H	427
287	BISYN, 2-D MULTI-GP DIFFUSION SYNTHESIS CALC	NED	635	F4	RS PL T	C	429
288	SNARG-1D, 1-D MULTI-GP DISCRETE ORDINATE CALC	ANL	3600	F36	RS P T	C	432
289	GAKER, INELASTIC SCAT X-SECTION CALC MODERATOR	GGA	1108	F4	RS P	A	434
290	GASA, STABILITY ANALYSIS REACTOR KINETICS EQNS	GGA	1108	F4	RS P	E	436
291	HEXSCAT, ELASTIC SCAT X-SECTIONS HEX LATTICE	GGA	1108	F4	RS	B	438
292	PSEUDO, STATISTICAL RESONANCE PARAMETER CALC	GGA	1108	F4	RS P	A	440
293	MARS, 2-D EXCURSION CALCULATION R-Z GEOMETRY	APDA	7094	F4	RS P	F	442
R294	MO89S,HOH, STEAM TABLES 14.5-2538 PSIA	ANL	1604	F63	RS P	H	444
R294	MO89S,HOH, STEAM TABLES 14.5-2538 PSIA	BAPL	6600	F4	RS P XT	H	444
R295	FLASH3, LOSS-OF-COOLANT ACCIDENT ANALYSIS	BAPL	6600	F4	RS P XT	G	446
296	GRDWRK, GRID GENERATION FOR SAFE PROGRAMS	GGA	1108	F4	RS P	L	448
297	CONTEMPT, LOSS-OF-COOLANT ACCIDENT ANALYSIS	PPCO	7040	F+MAP	RS P T	G	450
298	GGC4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA	GGA	1108	F4	RS PLXT	B	452
298	GGC4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA	GGA	6600	F4	RS L T	B	452
R299	LION, 3-D TEMPERATURE DISTRIBUTION PROGRAM	KAPL	6600	F+ASC	RS P T	H	456
300	SAFE-CREEP, VISCOELASTIC ANALYSIS CONCRETE	GGA	1108	F4	RS P	I	458
301	FREVAP6, HTGR METALLIC FISSION PRODUCT RELEASE	GGA	1108	F4	RS P	D	460
302	GAFFE, EQUILIBRIUM FUEL CYCLE CALCULATION	GGA	1108	F4	S T	D	462
303	BLOOST6, COMBINED KINETICS 2-D HEAT TRANSFER	GGA	1108	F4	RS P T	E	464
304	PERT4, 2-D PERTURBATION XY RZ RTHETA GEOMETRY	BNW	1108	F4	RS P	C	466
305	STRIP, RESOLVED RESONANCE INTEGRAL CALCULATION	BW	2000	F4	RS	A	468
306	FCC4, FUNDAMENTAL MODE FAST REACTOR X-SEC CALC	BNW	1108	F4	RS P T	B	470
307	HWOCR-SAFE, 2-D MONTE CARLO CELL CALCULATION	AI	360	F4	RS P XT	B	472
308	JUPITOR1, COUPLED-CHANNEL X-SEC EVALUATION	ORNL	1604	F63	RS P T	A	474
309	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS	AI	7094	F4	RS P T	F	476
309	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS	AI	360	F4	RS P T	F	476
310	GAKIN, 1-D MULTIGROUP TIME-DEPENDENT DIFFUSION	GGA	1108	F4	RS P T	F	478
311	BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS	GEV	635	F4	RS P T	O	480
312	TDSN, 2-D MULTIGROUP DISCRETE ORDINATE PROGRAM	LER	7090	F+MAP	RS P T	C	482
313	CINDER,MO102, POINT DEPLETION FISSION PRODUCT	BAPL	6600	F4	RS P	D	484
314	NAP, NEUTRON-INDUCED GAMMA-RAY RADIOACTIVITY	IITR	7094	F4	RSBPLXT	D	486
315	WIREX, COMPUTER-PRODUCED WIRING LISTS UHTREX	LASL	7090	F2	RS P	I	488
316	GAF,GAR, SPECTRA AND GROUP-AVERAGED X-SEC CALC	GGA	1108	F4	RS P XT	B	490
317	GAPCTKIN, SPACE-INDEPENDENT REACTOR KINETICS	GGA	1108	F4	RS P	E	493
318	FEVER7, 1-D MULTIGROUP DIFFUSION AND DEPLETION	GGA	1108	F4	RS P T	D	495
319	GASP7, 1-D BURNUP POWER DISTRIBUTION SEARCH	GGA	1108	F4	RS P T	C	497
320	TEMCC7, TEMPERATURE COEFFICIENT CALCULATION	GGA	1108	F4	RS P T	C	499
321	EXPALS, LEAST SQUARES EXPONENTIAL DECAY CURVES	LRL	7094	F2	RS P	P	501
322	ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC	ORNL	1604	F63	RS P	D	503
322	ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC	ORNL	360	F4	RS P	D	503
323	MUFFLE, FISSILE NUCLIDE X-SECTION EVALUATION	ORNL	7090	F4	RSBP	A	505
324	FRANTIC, LEAST SQUARES FIT SUM OF EXPONENTIALS	MIT	7090	F2	RS P	P	507
325	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	BNW	1108	F4	RS P T	D	509
325	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	LASL	6600	F4	RS P T	D	509
325	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	ANL	3600	F4	RSBP T	D	509
326	AIRCS2A, SIMULATION OF REACTOR DYNAMICS	AI	360	F+BAL	RS P T	E	511
R327	DAFT1, LEAST SQUARES FIT FISSILE NUCLIDE DATA	BAPL	6600	F4	RS P XT	P	514
328	NURLCC-1.0, LOSS-OF-COOLANT THERMAL ANALYSIS	BCL	6400	F4	RS P T	G	516
R329	MO457,PIPE, ELASTIC STRESS OF PIPING SYSTEM	BAPL	6600	F4	RS P XT	I	518
330	ECCSA1, LOSS-OF-COOLANT + EMERGENCY COOLING	BCL	6400	F4	RS P T	G	520
R331	FLOT1,MO219, PWR FLOW TRANSIENT ANALYSIS	BAPL	6600	F4	RS P XT	H	522

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333	TOAD, PROCESSING OF ANALYZER GAMMA-RAY SPECTRA	GGA	1108	F4	RS P	T O	526
334	PEGGY, ELASTIC SCATTERING PHASE-SHIFT ANALYSIS	ORNL	1604	F63	RS P	A	528
335	RAMES, PARTICLE WAVE FUNCTION RADIAL INTEGRALS	ORNL	1604	F63	RS P	A	530
R336	PDC5, 2-D FEW-GROUP DIFFUSION AND DEPLETION	IBM	360	F+BAL	RS P	T D	531
R337	STEM, MATRIX GENERATION FOR A SYSTEM OF BEAMS	KAPL	6600	F4	RS P	I	533
R338	TWIGGLE, 2-D 2-GP SPACE-TIME DIFFUSION	ANL	360	F4	RS P	F	535
R338	TWIGL, 2-D 2-GP SPACE-TIME DIFFUSION FEEDBACK	BAPL	6600	F4	RS P	XT F	535
339	GAUGE, 2-D FEW-GP HEX GEOM DIFFUSION DEPLETION	GGA	1108	F4	RS P	T D	538
340	POWERCO, NUCLEAR STATION ELECTRICITY COSTS	ORNL	1604	F63	RS P	D	541
341	GANDY, UNRESOLVED RESONANCE X-SECTION CALC	GGA	1108	F4	RS P	A	543
R342	MO648, 1-D SLAB TRANSPORT WITH SLOWING DOWN	BAPL	6600	F4	RS	XT C	546
R343	MO756,LETC, 1-D SLAB GAMMA-RAY TRANSPORT	BAPL	6600	F4	RS P	XT J	548
R344	GEM, EIGENVALUE PROBLEM FOR VIBRATING SYSTEMS	KAPL	6600	F4	RS P	T I	550
345	GAND, GAFGAR X-SECTION LIBRARY PREPARATION	GGA	1108	F4	RS P	T L	552
346	THTE, 3-D TRANSIENT HEAT TRANSFER PROGRAM	GEC	635	F+GMP	RSBP	T H	554
347	COCILLI, LEAST SQUARES ANALYSIS RESONANCE DATA	UILL	360	F4	RS P	A	556
R348	TOPS, TRANSIENT THERMODYNAMICS OF PRESSURIZERS	BAPL	6600	F4	RS P	XT H	558
349	TOCDEE, 2-D TIME-DEPENDENT HEAT CONDUCTION	PPCO	7044	F+MAP	RS P	T H	560
350	ETOE, ENDF/B TO MC**2 DATA CONVERSION	APDA	3600	ASAF4	RS P	T M	562
351	ECSIL, EXPERIMENTAL NEUTRON DATA LIBRARY	LRL	7094	F2	RS PL	T M	564
R352	RAUMZEIT, 1-D TIME-DEPENDENT DIFFUSION CALC	KAPL	6600	F4	RS P	F	567
353	SWELL2, FUEL ELEMENT LIFETIME ANALYSIS	ANL	3600	F36	RSBP	I	569
354	CINCAS, NUCLEAR FUEL CYCLE COST AND ECONOMICS	COMM	360	F4	RS P	T D	571
355	MC**2, ENDF MULTIGROUP X-SECTION CALCULATION	ANL	3600	F36	RS P	T B	573
356	ZPR-III ASSEMBLY 48 GAFGAR ENDF/B DATA TAPES	GGA	1108	BIN	R L	T Z	575
357	SUPORAN, REACTOR CORE SUPPORT STRESS ANALYSIS	ANL	3600	F36	RS P	T I	578
358	TWOTRAN, 2-D MULTI-GP TRANSPORT CODE X-Y GEOM	ANL	360	F4	RS P	T C	580
358	TWOTRAN, 2-D MULTI-GP TRANSPORT CODE R-Z GEOM	LASL	1108	F4	RS P	T C	580
358	TWOTRAN, 2-D MULTI-GP TRNSPT CODE XY RZ RTHETA	LASL	6600	F4	RS P	T C	580
R359	PUN1, UNRESOLVED RESONANCE INTEGRALS X-SECS	BAPL	6600	F4	RS P	XT A	582
360	TOR, THERMAL SCATTERING CRYSTALLINE MATERIALS	LASL	6600	F4	RS P	T A	584
361	GLEN, GROUP CONSTANT CALC FROM TOR OUTPUT DATA	LASL	6600	F4	RS P	T B	586
362	WELWING, MATERIAL BUCKLING CYL FUEL ELEMENTS	AEB	360	F4	S P	B	588
363	BLAST, REACTOR KINETICS TEMPERATURE DIST STUDY	AEB	360	F4	RS P	E	590
R364	SNEC, NONLINEAR ALGEBRAIC EQN SOLN CURVE PLOT	KAPL	6600	F+ASC	RS P	T P	592
365	BOW2, DEFLECTION CALCULATION PARALLEL BEAMS	ANL	3600	F36	SBP	I	594
366	CHEMLOC2, CORE HEATING CLADDING-STEAM REACTION	ANL	3600	F36	RSBP	G	596
367	ISOGEN, RADIONUCLIDE GENERATION AND DECAY	BNW	1108	F5	RSBPL	T D	598
368	FLANGE2, ENDF/B THERMAL SCATTERING DATA PROC	DP	360	F4	RS P	T A	600
368	FLANGE2, ENDF/B THERMAL SCATTERING DATA PROC	GGA	1108	F4	RS	T A	600
369	RELAP2, REACTOR BLOWDOWN - EXCURSION ANALYSIS	INC	7044	F4	RS P	T G	602
370	GAKIT, 1-D MULTIGP KINETICS WITH TEMP FEEDBACK	GGA	1108	F4	RS P	T F	604
R371	NOWIG, 1-D 2-GP KINETICS TEMPERATURE FEEDBACK	BAPL	6600	F4	RS P	XT F	607
372	RAPFU, FUEL CYCLE PARAMETERS FAST BREEDERS	APO	635	F4	RS P	D	609
R373	BL47, DRAFTING TOOL TO PLOT PLANE STRUCTURES	BAPL	6600	F4	RS P	XT L	611
374	1DX, 1-D DIFFUSION FAST X-SECTION GENERATION	BNW	1108	F4	RS P	XT B	613
375	SCORE2, SCISRS ENDF/B GRAPHIC X-SEC EVALUATION	AI	360	F+BAL	RS L	T M	615
376	AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC	BNL	6600	F4	RS P	A	617
376	AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC	BNL	7094	F4	RS P	A	617
377	SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC	BNL	6600	F4	RS P	A	619
377	SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC	BNL	7094	F4	RS P	A	619
R378	TUBE, U-TUBE HEAT EXCHANGER STRESS ANALYSIS	KAPL	6600	F4	RS P	I	621
379	SAFE-2D, PLANE + AXISYMMETRIC STRESS ANALYSIS	GGA	1108	F4	RS P	I	623
380	GATT, 3-D FEW-GP DIFFUSION CALC HEX-Z MESH	GGA	1108	F4	RS P	T C	625
381	LYNNE, WOODS-SAXON POTENTIAL SHAPE CALCULATION	ORNL	7090	F2	RS	A	627
R382	RAPP, HIGH-VELOCITY FLOW STUDY STEAM-WATER MIX	KAPL	6600	F4	RS P	H	628
R383	MO266, LINEAR ELASTIC STRUCTURAL DYNAMICS	BAPL	6600	F4	RS P	XT I	630
384	CHECKER,CRECT,CAMMET,PLOTFB,SLAV3, ENDF/B PROC	NCSC	6600	F4	RS	T M	632
385	COHBE, COHERENT INELASTIC SCATTERING LAW CALC	GGA	1108	F4	RS P X	A	635
R386	DATATRAN, MODULAR PROGRAMMING AND DATA SYSTEM	KAPL	6600	F+COM	RSB	T M	637
387	CITATION, 1,2,3-D DIFFUSION DEPLETION MULTIGP	ORNL	360	F4	RS P	XT K	639
388	ETOX, MULTIGROUP CONSTANTS FROM ENDF/B FOR 1DX	BNWL	1108	F4	RS P	T B	642
R389	STINT3, SINGLE-CHANNEL SPACE-TIME SYNTHESIS	KAPL	6600	F4	RS P	F	644
390	CORGAM, UNFOLDING OF COMPLEX GAMMA-RAY SPECTRA	KSUN	360	F4	RS P	O	646
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392	RAFFLE, 1ST FLIGHT COLLISICN PROBABILITIES MC	ORNL	7090	F+FAP	RS P	T B	650
393	XSDRN, DISCRETE ORDINATE MULTIGROUP CONSTANTS	ORNL	360	F4	RS PLXT	B	652
R394	JITER, FLUCTUATION EXPERIMENT ANALYSIS	BAPL	6600	F4	RS P	T O	654
395	FLAC, STEADY-STATE FLOW, PRESSURE DISTRIBUTION	GGA	1108	F5	RS P	H	656
R396	WASP, WATER AND STEAM THERMODYNAMIC PROPERTIES	BAPL	6600	F4	RS P	XT H	658
R397	GAPL3, INELASTIC LARGE DEFLECTICN STRESS STUDY	BAPL	6600	F+COM	RS P	XT I	660
R398	BE21, FEW-GP DISCRETE ORDINATES SLAB GEOMETRY	BAPL	6600	F4	RS P	XT C	662
R399	SUMCR, S-WAVE NEUTRON X-SECTION CALCULATION	BAPL	6600	F4	RS P	XT A	664
400	SAS1A, FAST REACTOR POWER AND FLOW TRANSIENTS	ANL	360	F4	RS P	T F	666
400	SAS1A, FAST REACTOR POWER AND FLOW TRANSIENTS	ANL	6600	F4	RS P	T F	666
401	GAMTRI, 2-D MULTIGP DIFFUSION TRIANGULAR MESH	GGA	1108	F+BAL	RS P	T C	668
R402	SABOR4, DISCRETE-ELEMENT ANALYSIS THIN SHELLS	MIT	360	F4	RS P	T I	670
R403	TIGIR2, MODULAR DOCUMENT INFORMATION SYSTEM	KAPL	6600	F	R B	T M	673
R404	FINEL, FINITE-ELEMENT STUDY 2,3-D STRUCTURES	KAPL	6600	F4	RS P X	I	675
405	NOAH, 1-D ONE-GP SPACE-TIME DIFFUSION FEEDBACK	BNL	6600	F4	RS P	T F	677
R406	DATATRAN 2-D GEOMETRY INPUT, PREPARATION, EDIT	KAPL	6600	F+ASC	RS P	T L	679
R407	DATATRAN UTILITY MODULES, 2-D + 3-D PLOTTING	KAPL	6600	F+ASC	RS P	T P	681
408	TAC2D, STEADY-STATE AND TRANSIENT TEMP CALC	GGA	1108	F5	RS P	T H	683
409	LARCA, FLUX-WEIGHTING OF DTF4 CROSS SECTIONS	LASL	6600	F4	RS P	N	685
410	TACASI, ANALYSIS OF RESCNANCE MEASUREMENTS	GGA	1108	F4	RS P	T A	687
R411	M0661,M0657,M0626, POLYNCMIAL CURVE FITTING	BAPL	6600	F4	RS P	XT P	689
R412	MANE1, RECTANGULAR MAGNETIC NETWORK SOLUTION	BAPL	6600	F4	RS P	XT I	691
413	ALPHA-M, RESOLUTION OF GAMMA RAY SPECTRA	ORNL	360	F4	RS P	O	693
414	TAC3D, TRANSIENT 3-D HEAT TRANSFER PROGRAM	GGA	1108	F5	RS P	T H	695
415	CEXE,INCEXE, 1-GP 3-D XYZ XENCN OSCILLATION	CE	360	F4	RS P	XT F	697
416	PARTI, OPTIMAL GROUP OR MESH CCLLAPSING	TRW-MMU	6500	F4	RS P	B	699
417	ATHENA4, INELASTIC SCATTERING FORM FACTORS	ORNL	360	F4	RS P	T A	701
418	CHAINS, ANALYSIS OF RADIOACTIVE DECAY CHAINS	GEC	635	F4	RSBP	D	703
419	CASCADE,CLUSTER, RADIATICN DAMAGE IN METALS	GEC	635	F+FAP	RSBP	T Q	705
420	GRCLSE, SPACE-DEPENDENT X-SECTICN GENERATION	GEC	635	F4	RSBP	T B	707
421	MICHRD, MICROHARDNESS MEASUREMENT ANALYSIS	GEC	635	F4	RSBP	O	709
422	SPECTRA, MC CALCULATION IRRACIATED MATERIAL	GEC	7090	FAP	RSBP	XT Q	711
423	DOS, NEUTRON FLUX-DOSIMETER ACTIVITY RELATION	GEC	635	F4	RSBP	D	713
R424	GLUB1, WATER-LOGGED FUEL ELEMENT ANALYSIS	BAPL	6600	F4	RS P	XT H	715
R425	REDUX, REACTOR FLUCTUATION EXPERIMENT ANALYSIS	BAPL	6600	F4	RS P	XT O	717
R426	PAXO2, HARMONY-PDQ X-SECTICN GENERATION CODE	BAPL	6600	F4	RS P	XT B	719
427	FAREC, 1-D FAST REACTOR DESIGN & SURVEY STUDY	B&W	6600	F+CCM	RS P	XT D	721
R428	DOGGY, DESK CALCULATOR FORM SHEET DP PACKAGE	KAPL	6600	F+ASC	RS P	T P	725
R429	ASPIS, GAMMA RAY SOURCE BUILDUP FACTOR CALC	BAPL	6600	F+ASC	RS P	XT J	727
430	GAZE2, 1-D MULTIGROUP DIFFUSICN SLAB, SPH, CYL	GGA	1108	F4	RS P	T C	729
431	SUPERTOG, ENDF/B FINE-GP CCNSTANTS GENERATION	ORNL	360	F4	RS P	T B	731
432	COBRA2, ROD BUNDLE THERMALHYDRAULIC ANALYSIS	BNWL	1108	F4	RS P	T H	733
432	COBRA2, ROD BUNDLE THERMALHYDRAULIC ANALYSIS	AJ-ANL	360	F4	RS P	T H	733
433	CONTEMPT,CONPS, PRESSURE-TEMPERATURE RESPONSE	INC	360	F+BAL	RSBP	T G	735
434	HEATMESH, GEOMETRICAL DATA HEAT TRANSFER STUDY	SLL	6600	F4	RS P	N	738
435	BURST1, HYDRODYNAMIC ANALYSIS DURING BLOWDOWN	PPCO	360	F4	RS P X	G	740
436	ETCM1, ENDF/B FORMAT TO MUFT FORMAT X-SECTIONS	WNES	6600	F4	RS P	T B	742
437	ETCG1, ENDF/B TO MUFT, GAM, ANISN X-SEC FORMAT	WNES	6600	F4	RS PL	T B	744
438	BUG2, 2-D MULTIGROUP DIFFUSICN & BURNUP XY, RZ	GGA	1108	F4	RS P	T D	747
439	BUGTRI, 2-D MULTIGP DIFFUSICN+BURNUP TRI-MESH	GGA	1108	F4	RS P	T D	749
440	DYNAM, DYNAMIC ANALYSIS BOILING FLOW STEAM	GGA	1108	F4	RS P	T H	751
441	PWCOST, REACTOR FUEL CYCLE COST CALCULATION	GGA	1108	F4	RS P	T D	753
R442	SIMPLE1, TIME-SHARING PROGRAMMING LANGUAGE	KAPL	6600	F+ASC	RS	P	755
443	HAA3, CCAGULATICN OF HETERCGENECUS AEROSOLS	AI	360	F4	RS P	T G	757
R444	ROPE, FINDING ROOTS OF A PCLYNCMIAL	KAPL	6600	F4	RS P	P	759
R445	LIZARD, NONLINEAR DIFFERENTIAL EQNS. SOLUTION	KAPL	6600	F+ASC	RS P	T P	761
R446	MOST, A MULTIDIMENSIONAL OPTIMIZATION SCHEME	KAPL	6600	F4	RS P	P	763
447	ETCG1 DATA LIBRARIES, MUFT4 OR 5 + GAM1 + GAM2	WNES	6600	BCD	R L	T Z	764
R448	FLASH4, FULLY-IMPLICIT TRANSIENT SIMULATION	BAPL	6600	F4	RS P	XT G	766
R449	CYGRO3, OXIDE FUEL ROD STRESS & DEFORMATION	BAPL	6600	F4	RS P	XT I	768

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ARGONNE CODE CENTER: COMPILATION OF PROGRAM ABSTRACTS

by

M. K. Butler, Pamela Henline,
Marianne Legan, L. Ranzini,
and William J. Snow

PREFACE

This is the second complete revision of program abstracts undertaken by the Code Center since its inception in 1960. This revision has eliminated from our collection those programs for machines of the 704 era and earlier. With the assistance of installation representatives and authors, existing program abstracts have been revised and material added to the library to complete, whenever possible, the program package for each program in the Center's collection.

This is the first attempt at a formal publication. Previous abstracts have been processed by use of multilith masters from high-speed printers. Publishing this as an AEC document makes it more easily referenced and accessed. Additions and corrections will be published as previously, with provision for a complete revision in this form after another 3- to 5-year period.

The table of contents lists in order of accession, the KWIC title of each program, the originating installation, the machine, the programming language, the package symbols, and the category. This title information was then used to provide the index which follows the abstracts. The package symbols included are defined as follows:

R = reference report or document,
S = source deck or tape,
B = binary or run deck or tape,
P = sample problem data,
L = library information, e.g., cross-section library, random-number library,
X = auxiliary routines,

and

T = magnetic tape(s) required for transmittal of card and/or tape material.

This collection includes programs in which the program number is preceded by the letter R. This denotes restricted, and U.S. citizens are

required to file a release form with the Center for use of the program on either official U.S. government or commercial (non-military) applications at a U.S. installation. The Code Center is not authorized to distribute these programs outside the United States.

To best service the interests of AEC contractors, university nuclear-engineering departments, and the nuclear industry, the Center will, upon request, supply to an installation, a copy of any of the material listed as available under Item 16 of the abstract. Should that installation modify the program, convert it to a usable form on another computer, uncover an error, or have difficulty in using the program, he is asked to notify the Center so that his efforts in turn might benefit other recipients. The Center maintains records of the recipients of all library material so that if a correction or new version is prepared, all previous recipients of the program are notified promptly.

Program abstracts for the European Nuclear Energy Agency Computer Programme Library programs in the Argonne Code Center collection have not been included in this compilation. Separate compilations are distributed regularly to our contributing installations.

I. HISTORY AND ACKNOWLEDGMENTS

Almost twelve years ago, at the Second Annual Meeting of the American Nuclear Society, the initial meeting of an informal group called the Nuclear Codes Group was held. This group was composed of "persons interested in the development and use of computer programs for reactor design." The group held regular meetings in conjunction with the semi-annual ANS meetings and published and distributed a Newsletter containing code abstracts submitted by members. These abstracts followed the format of the Atomic Energy Commission document AECU-3078, "A Bibliography of Available Digital Computer Codes for Nuclear Reactor Problems."[1]

From September 1956 to December 1959, ten Newsletters were published and distributed through the auspices of the AEC Computing Facility at New York University.[2] In 1959, Ward Sangren and personnel at General Atomics edited and published 300 abstracts collected from Group members.[3] A supplementary list of 100 abstracts was published the following year.[4]

In 1959 also, the Nuclear Codes Group became the nucleus of the American Nuclear Society's Mathematics and Computation Division. With stature came a desire for a central facility for the computer program library and the dissemination of information concerning programs written in the areas of nuclear physics, reactor design, and engineering. In 1960,

a proposal was made whereby, with the cooperation of Division members and installation representatives, and the Atomic Energy Commission's approval, Argonne National Laboratory undertook to serve as this central agency and the Argonne Code Center was established.

Procedures were drafted defining the material to be collected--the abstract and the "package"--and the responsibilities of the Center and the Installation Representatives. The submission of the material is the responsibility of the authors either directly, or through their Installation Representative. The Center edits and publishes the abstracts, and collects, maintains, tests, and distributes the packages. The representatives at AEC laboratories, contracting establishments, and universities serve as a source of information to the Center concerning programs or requests emanating from their installation, and it is they, in collaboration with the personnel at the Argonne Center, who supply the answers to the questions and nonroutine inquiries received. They are responsible for library programs at their installations.

In 1965 the Argonne Center became a separately-budgeted operation of the Reactor Physics Branch of the USAEC Division of Reactor Development and Technology, which is responsible for support of its current program.

From 1961 until issuance of this publication in January 1968, the Argonne Code Center published 12 distributions containing abstracts of approximately 265 programs.[5] This collection represents a complete revision of those previously published abstracts and includes, in addition, abstracts prepared from 1968 to date.

Since the Code Center has been in operation, other specialized program libraries and computer-code collections have been established. These include our cooperating European counterpart, the ENEA Computer Programme Library, established at Ispra, Italy, and the Radiation Shielding Information Center at Oak Ridge National Laboratory (both established in 1964), and the National Aeronautics and Space Administration COSMIC activity, initiated at the University of Georgia in 1966.

II. ABSTRACT FORMAT

The program abstract has been modified considerably over the years. Its evolution from the original AECU-3078 format is due primarily to our experience at the Center and cooperative efforts undertaken with the ENEA library personnel and the American Nuclear Society's ANS-10 subcommittee membership.[6]

A. Name or Designation of Program

This is the name or designation given the program(s) by the author.

B. Computer for Which Program Is Designed and Others upon Which It Is Operable

This is the name of the computer for which the original program was written, and, in addition, names of any other computers for which the program has been converted and entered into the library even though minor modifications may have been made.

C. Nature of Physical Problem Solved

This is a brief description of the physical problem, including any basic physics approximations involved in the problem formulation.

D. Method of Solution

This is a short summary of the mathematical and numerical techniques or algorithms used in the calculation.

E. Restrictions on the Complexity of the Problem

This item includes restrictions implied by storage allocation, such as maximum number of energy groups and mesh points, as well as implied argument-range restrictions due to approximations used, etc.

F. Typical Running Time

This is the detailed information needed to enable the potential user to estimate the running time for a given choice of program parameters.

G. Unusual Features of the Program

This states the distinguishing features of this program and the class of problems that most effectively benefit therefrom. This category should allow the user to select, from a number of similar programs, the one most suitable for a particular problem.

H. Related and Auxiliary Programs

The details should be given here if this program supersedes or is an extension of an earlier program and if other programs are used in connection with this program, either for processing input or output or as subprograms.

I. Status

The Center lists here the initial date of publication of the abstract, as well as the dates on which each version in the library was submitted. Any testing completed at the Center is noted.

J. References

This lists generally available material and documentation pertinent to the program.

K. Machine Requirements

This lists all hardware components necessary for full utilization of the program, including such items as the amount of high-speed memory, any auxiliary or backup storage, such as tapes, disks, and drums, the channel(s) configuration, and other input-output equipment such as on-line punch, printer, clock, display, or recorder.

L. Programming Language(s) Used

This states the programming language or languages in which the program has been written and indicates what percentage is in each. If a certain class of routines is in assembly rather than compiler language, this should be mentioned (in particular, the input-output routines).

M. Operating System or Monitor under Which Program Is Executed

Based on the operating system or monitor and associated subroutine library distributed by the computer manufacturer as "standard," this summarizes deviations pertinent to the operation of the program. It also indicates the installation environment report, if any, associated with the program.

N. Other Programming or Operating Information or Restrictions

This summarizes additional information necessary to permit the reader to decide the extent of changes necessary to convert this program to his use in another operating environment--operating system, programming language, and/or computer.

O. Name and Establishment of Author

This information refers to initial submission. Normally the author will be author of both the program and the abstract. Authors of different computer versions will be indicated here as well.

P. Material Available

This lists the material being distributed, i.e., the program package contents.

Q. Category

This is the problem classification assigned from the Center program classification guide.

Keywords

This is a listing of the keywords associated with the program, supplied by the program author and/or the Center based on the Argonne Code Center thesaurus.

III. RECOMMENDED PROGRAM PACKAGE CONTENTS

This section has been revised to conform to the recommendations outlined in the ANS publication, Nuclear Engineering Bulletin 4-1, September 1966, entitled "A Code of Good Practices for the Documentation of Digital Computer Programs."

A. Card and/or Tape Material

1. Source Deck

These are the compiler and/or assembly language programs as punched on cards or recorded in card image form on magnetic tape.

2. Run Deck

This is the program object deck, including any nonstandard library subroutines utilized, and operating system or monitor control cards set up for execution of the sample problem.

3. Sample Problem

This is the input data deck as set up for execution. Machine listing of storage allocation map and the problem output recorded when executing this sample problem are included if not available in the documentation.

4. Data Library

This includes all data files required for program operation, e.g., cross sections, steam tables.

5. Auxiliary Routines

These are any subsidiary programs useful in preparing input information, processing results, maintaining data libraries, etc.

B. Program Report(s)

1. Program Description

This is the definition of the physical problem as well as the mathematical model. The mathematical and numerical methods employed are described.

2. User's Information

This is the complete set of information required to use the program effectively. It includes operating instructions and a description of the input and output formats and program options in sufficient detail to enable the user to specify his own problem and interpret the results. Input and selected output for a sample problem are included as an example. This example is identical to the sample problem included in the card and/or tape material of the program package.

3. Programmer's Information

Special documentation is provided addressed to program modification and the problem of transferring the program to another machine. Sufficiently detailed information is provided to permit a programmer from another installation to decide whether he can use the program on his computer and then expedite the modifications necessary to make the program operational. This includes:

a. A description of the hardware configuration employed, including the memory heirarchy and size, and the program utilization of each component.

b. Operating system information, including loader facilities, restrictions, and limitations, especially in regard to segmented programs.

c. Flow charts and descriptions of the performance of machine-dependent routines.

d. Procedures for use and maintenance of auxiliary data libraries required for effective use of the program.

e. Glossaries giving the correspondence between the symbols used in the text of the report and those used in the program.

f. An overall view of the logical flow of the program either in schematic flow-chart or descriptive form.

The detailed environment information, items a-d above, can be provided in two ways. One, it may be included in the program report and

should then be consistent with the description of the Installation Environment Report given in Section C below, as pertinent to this program. Or, two, reference may be made to a particular Installation Environment Report with only those optional features and/or deviations pertinent to this program noted in the Program Report.

C. Installation Environment Report

Although a computer program may be completely written in machine-independent language, such as FORTRAN, the program requires a certain minimum environment to operate properly. This environment consists of all the software and hardware devices required to compile and execute the program.

This section of the documentation is designed to specify this required environment. It should be adequate for specification of a replacement environment at another installation. As a minimum, each unit of hardware and software assumed should be listed with a description of how each device is used and how each routine functions. Reference should be made to a complete description, when generally available.

The hardware environment is simply the total collection of devices used either directly or indirectly by this program. This is meant to include mass storage devices, on-line typewriters, clocks, sense switches, and various input and display devices. The amount and hierarchy of memory required for data storage are an important attribute of the hardware environment.

The software environment may be considered in three parts. The first part is the standard set which may be expected to exist at any installation and is generally provided by the manufacturer. This includes standard library subroutines, such as sin and cos; a loader for relocatable loading and linking; a standard compiler for the language use; and an operating or control system. The second part is a set of special configuration-tailored subroutines for data manipulation, storage, and retrieval used in place of statements such as "read" and "write." With the rapid proliferation of mass storage devices, such subroutines can be expected to grow in number and variety. The third part is a catchall category for all other software, such as special matrix manipulative subroutines, free-format input routines, and plotter subroutines, which the originating installation uses as a matter of course in producing many programs.

It is recommended that each installation provide general Installation Environment Reports that can be referenced for details and exceptions in the individual program reports. Although these environment reports are subject to change and must be updated periodically and uniquely identified to serve the reference purposes proposed, their use can greatly simplify the preparation of program packages and facilitate effective program exchange.

IV. PROGRAM CLASSIFICATION GUIDE AND THESAURUS

A program classification scheme is being initiated with this report. It is intended to serve as a guide to those library programs dealing with the solution of a specific physical problem or area of related problems. In addition, a thesaurus has been prepared as a basis for the assignment of keywords to the library programs. The category and keywords listed as Abstract Item 17 have been assigned by the Code Center staff, but it is hoped that in the future they will be selected from the guide and thesaurus by the author of the program and/or abstract. The thesaurus follows the concept and design of the EURATOM thesaurus[7] insofar as practicable. Accepted keywords not included in the thesaurus include specific reactor and computer code names that are always followed by the word "reactors" or "codes," e.g., HTGR reactors, AITP3 codes. Other accepted keywords are the element names and combinations of an element name and a mass number, e.g., hydrogen, uranium-238.

Program Classification Guide

A. Cross-section and Resonance-integral Calculations

Computation of reaction cross sections from nuclear theory such as the optical or Hauser-Feshbach models, resonance cross sections by Breit-Wigner or multilevel theory, determination of differential scattering cross sections, cross-section evaluation, and compilation programs.

B. Spectrum Calculations, Generation of Group Constants, Lattice and Cell Problems

Determination of the slowing-down density or thermal spectrum, weighting and averaging of cross sections and related quantities for the production of group constants, and evaluation of design parameters by lattice and cell calculation.

C. Static Design Studies

Calculation of the reactivity and flux distribution of the reactor system, and adjustment of design parameters to prescribed specifications, i.e., criticality and power distribution search procedures.

D. Depletion, Fuel Management, Cost Analysis, and Reactor Economics

Includes burnup programs, isotope and fission-product buildup and decay computations, and optimization studies.

E. Space-independent Kinetics

Studies of the time-behavior of reactors, including delayed-neutron effects and feedback mechanisms, and transfer-function evaluation.

F. Space-Time Kinetics, Coupled Neutronics-Hydrodynamics-Thermodynamics and Excursion Simulations

Programs that consider spatial design characteristics and accompanying effects in studying the time behavior of the reactor.

G. Radiological Safety, Hazard and Accident Analysis

Calculation of internal and external dose rates, determination of reactor thermodynamic and hydrodynamic properties following an accident, e.g., release of radioactive materials, coolant system blowdown, steam generator rupture.

H. Steady-state and Transient Heat Transfer

Includes fluid-flow studies and calculations of thermodynamic properties.

I. Deformation and Stress Distribution Computations, Structural Analysis, and Engineering Design Studies

Includes fuel-element design evaluations, core-configuration studies, and composite structure analysis.

J. Gamma Heating and Shield Design Programs

Computation of heat-generation rates, and penetration analysis and leakage calculations for reactor shields.

K. Total Systems Analysis

Collections of solutions to correlated problems elicited from several categories, designed and used as systems.

L. Data Preparation

Generation of program parameters; checking, editing, and formatting of problem input information.

M. Data Management

Construction, maintenance, and retrieval of data files, e.g., cross-section libraries.

N. Subsidiary Calculations

Plotting, editing, and display routines that process output data from other programs.

O. Experimental Data Processing

Programs designed to process data directly acquired from an experimental situation or to assist the experimenter in the design of the experiment, including instrument response and correction factor calculations.

P. General Mathematical and Computing System Routines

Calculation of mathematical functions, and special-language routines with general data-processing capabilities.

Q. Radiation Effects

Simulation of radiation damage processes in metals.

Z. Nuclear Data

Data prepared in specified program formats for benchmark studies, program testing, etc.

Thesaurus

1-dimensional
1-group
2-dimensional
2-group
3-dimensional
absorption
accidents
activation
aerosols
age
alpha decay
angular distribution
anisotropic scattering
annular space
assembler
atmosphere
averages
B_L method
background
beams
blowdown
Breit-Wigner formula
breeding
Brown-St. John model
buckling
capture
cell calculation
coefficients
Cohen equation
coincidence methods
compound nuclei
containment
continuous release
control
control rods
coolants
correlation
criticality
cross sections
crystals
currents
cylinders
Dancoff correction
data processing
decay
deformation
depletion
design
differential equations
diffusion
disadvantage factors
distance
distribution
Doppler broadening
Doppler coefficient
dose rates
dynamics
economics
efficiency
elastic
elasticity
enthalpy
epithermal
equations
excursions
expansion
factor
failures
fast
feedback
few-group
finite-element

fission
fission products
fluid flow
fluids
flux
Fourier transform
frequency
fuel cycle
fuel elements
fuels
gamma radiation
gas coolants
gases
geometries
graphs
group constants
Hauser-Feshbach theory
heat conduction
heat transfer
heating
heavy
heterogeneous
hexagonal
homogeneous
hydrodynamics
incoherent approximation
inelastic
infinite media
input data
instantaneous release
isotopes
isotropic scattering
kinetics
lattices
leakage
least squares
Legendre coefficients
libraries
lifetime
light
liquids
liquid metals
magnetic
maintenance
mass matrices
Maxwell distribution
measurements
minimization
moderators
Monte Carlo method
multidimensional
multigroup
multilevel
neutrons
nonlinear
numerical calculation
operation
optical model
output data
parameters
particles
performance
perturbation theory
pipes
P_L method
plates
poison
polynomials
potential scattering
power
power plants
preparation
pressure
production
programming languages
pumps
r-theta
r-z
radiation effects
radioactivity
reaction rates
reactivity
reactor safety
reactors
resolved region
resonance
resonance escape probability
resonance integrals
retrieval
scattering
scattering law
scintillation counters
searches
Selengut-Goertzel equation
shells
shielding
slabs
slowing down
S_n method
solids
solutions
space
space-independent
space-time
spectra
spheres
statistics
steam generators
stresses
structural analysis
swelling
synthesis
temperature
temperature coefficient
thermal
thermal utilization
thermalization
thermodynamics
transfer functions
transients
transport theory
triangular
unresolved region
vapors
variations
velocity
vibrations
water
weight
Wigner-Wilkins model
Wilkins equation
x-y
x-y-z

15. NAME AND ESTABLISHMENT OF AUTHORS -

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ARGONNE NATIONAL LABORATORY
P. O. BOX 2528
IDAHO FALLS, IDAHO 83401

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (6600 VERSION)
SOURCE DECKS (6600-1022 CARDS, 1604-965 CARDS)
SAMPLE PROBLEMS (6600-7 CARDS, 1604-7 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (6600-21,125 CARDS)
REFERENCE REPORTS, WAPD-TM-680 AND WAPD-TM-668, AND NOTE

17. CATEGORY - H
KEYWORDS - THERMODYNAMICS, WATER, PRESSURE, TEMPERATURE, LIQUIDS, VAPORS

REACTOR CODE ABSTRACT 295 01/71

1. NAME OR DESIGNATION OF PROGRAM - FLASH3

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - FLASH3 DETERMINES THE TRANSIENT RESPONSE OF A WATER-COOLED 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 - FLASH3 CAN BE APPLIED TO ANY WATER-COOLED REACTOR GEOMETRY OR PLANT CONFIGURATION. IT INCLUDES ACOUSTIC EFFECTS FOR BOTH SUBCOOLED AND SATURATED BLOWDOWNS.

8. RELATED AND AUXILIARY PROGRAMS - FLASH3 IS AN EXTENSION OF FLASH AND FLASH2 AND USES THE INPUT PROCESSING SUBROUTINES FCHIP, CARDS, AND INPF (SEE REFERENCE 4).

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1967.
CDC6600 VERSION OF FLASH2 SUBMITTED SEPTEMBER 1967,
REPLACED BY CDC6600 VERSION OF FLASH3 JANUARY 1970.

10. REFERENCES - J. H. MURPHY, J. A. REDFIELD, AND V. C. DAVIS, FLASH-3 A FORTRAN IV PROGRAM FOR THE SIMULATION OF REACTOR PLANT TRANSIENTS IN SPACE AND TIME, WAPD-TM-800, JULY 1968.
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 3.1.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (10,806 CARDS)
SAMPLE PROBLEM (61 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS, WAPD-TM-800 AND WAPD-TM-668

17. CATEGORY - G
KEYWORDS - ACCIDENT, CONTAINMENT, REACTORS, FUELS, COOLANTS, FLUID FLOW, PRESSURE DISTRIBUTION, TEMPERATURE DISTRIBUTION, BLOWDOWN, REACTOR SAFETY

REACTOR CODE ABSTRACT 296 12/67

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-PORS (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-PORS, 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.

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

REACTOR CODE ABSTRACT 329 01/71

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-TRAILS
   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, REPLACED BY UPDATED CDC6600 VERSION MAY 1970.

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 -

15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (2174 CARDS)
SAMPLE PROBLEM (50 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS

17. CATEGORY - I
KEYWORDS - ELASTICITY, THERMAL STRESSES, PIPES, 3-DIMENSIONAL, PRESSURE

REACTOR CODE ABSTRACT 330 6/68

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 - 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 - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (2230 CARDS)
SAMPLE PROBLEM (25 CARDS)
REFERENCE REPORT AND SUPPLEMENT

17. CATEGORY - I
KEYWORDS - STRUCTURAL ANALYSIS, STRESSES, DEFORMATION

REACTOR CODE ABSTRACT 358 01/71

1. NAME OR DESIGNATION OF PROGRAM - TWOTRAN

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108, CDC6600, IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - TWOTRAN SOLVES TWO-DIMENSIONAL PARTICLE TRANSPORT PROBLEMS. SEPARATE VERSIONS ARE AVAILABLE IN X-Y AND R-Z GEOMETRIES AS WELL AS A GENERAL GEOMETRY VERSION IN X-Y, R-Z, AND R-THETA GEOMETRIES. BOTH DIRECT AND ADJOINT, HOMOGENEOUS (KEFF OR PARAMETRIC EIGENVALUE SEARCHES) OR INHOMOGENEOUS TIME-INDEPENDENT PROBLEMS ARE SOLVED SUBJECT TO VACUUM, REFLECTIVE, OR INPUT SPECIFICATION OF BOUNDARY FLUX CONDITIONS. BOTH ANISOTROPIC INHOMOGENEOUS PROBLEMS AND GENERAL ANISOTROPIC SCATTERING PROBLEMS ARE TREATED.

4. METHOD OF SOLUTION - ENERGY DEPENDENCE IS TREATED BY THE MULTI-GROUP APPROXIMATION AND THE ANGULAR DEPENDENCE BY A DISCRETE ORDINATES APPROXIMATION. IN THE X-Y VERSION SPACE DEPENDENCE IS APPROXIMATED BY A POSITIVE, WEIGHTED, DIAMOND DIFFERENCE SCHEME OR (AT THE USERS OPTION) BY THE SOMEWHAT MORE ACCURATE (BUT NOT POSITIVE) EQUAL WEIGHT DIAMOND DIFFERENCE SCHEME. IN THE GENERAL GEOMETRY VERSION SPACE DEPENDENCE IS APPROXIMATED BY THE DIAMOND DIFFERENCE SCHEME WITH A SET-TO-ZERO NEGATIVE FLUX CONTROL. IN BOTH VERSIONS ANISOTROPIC SCATTERING AND ANISOTROPIC INHOMOGENEOUS SOURCES ARE REPRESENTED BY FINITE SPHERICAL HARMONICS EXPANSIONS. WITHIN-GROUP ITERATIONS, UPSCATTERING ITERATIONS, KEFF ITERATIONS, AND EIGENVALUE SEARCH ITERATIONS ARE ACCELERATED BY A COARSE-MESH PARTICLE REBALANCING ALGORITHM, AND, IN THE X-Y VERSION AT THE OPTION OF THE USER, WITHIN-GROUP ITERATIONS INCLUDING IMPLICIT BOUNDARY CONDITIONS ARE ACCELERATED BY AN AUTOMATIC CHEBYSHEV ACCELERATION.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE VARIABLE DIMENSIONING CAPABILITY OF FORTRAN IV IS USED SO THAT ANY COMBINATION OF PROBLEM PARAMETERS LEADING TO A BLANK COMMON VECTOR LENGTH LESS THAN MAX IN THE X-Y VERSION (LENXCA IN THE GENERAL GEOMETRY VERSION) CAN BE USED. IN THE X-Y VERSION FOR A 65,536 WORD MACHINE MAX IS SLIGHTLY GREATER THAN 30,000 WORDS. IN THE GENERAL GEOMETRY VERSION FOR A 65K MACHINE LENXCA CAN BE GREATER THAN 34,000 DEPENDING ON LOCAL SYSTEM REQUIREMENTS. WITH A FEW EXCEPTIONS, ONLY WITHIN-GROUP PROBLEM DATA ARE STORED IN FAST MEMORY AND DATA FOR ALL OTHER GROUPS ARE STORED IN AUXILIARY STORAGE. ARBITRARY NUMBERS OF GROUPS OF UP OR DOWN SCATTERING ARE ALLOWED.

6. TYPICAL RUNNING TIME - FOR THE X-Y VERSION ON A CDC6600 A 7-GROUP, LINEAR ANISOTROPIC, UPSCATTERING, 40 X 40 S4 KEFF CALCULATION REQUIRES 57.23 MINUTES. A 1-GROUP, ISOTROPIC SCATTERING, 32 X 32 S12 INHOMOGENEOUS SOURCE CALCULATION REQUIRES 0.54 MINUTE (PURE ABSORBER) OR 1.81 MINUTES (C=0.99). 1.26 MINUTES ARE REQUIRED BY A 2-GROUP, LINEAR ANISOTROPIC, 10 X 10 S6 CRITICAL DIMENSION CALCULATION WITH BOTH X AND Y DIMENSIONS MODIFIED. IN THE GENERAL GEOMETRY VERSION AN X-Y, 7-GROUP, LINEAR ANISOTROPIC, UPSCATTERING

6. TYPICAL RUNNING TIME (CONTINUED)
40 X 40, S4 KEFF CALCULATION REQUIRES 64.73 MINUTES. AN R-THETA, 9-GROUP, LINEAR ANISOTROPIC, UPSCATTERING 42 X 10, S4 KEFF CALCULATION REQUIRES 12.83 MINUTES. AN R-Z, 10-GROUP, P2 SCATTERING, 21 X 20, S8, SOURCE PROBLEM REQUIRES 21.34 MINUTES.

7. UNUSUAL FEATURES OF THE PROGRAM - UNUSUAL FEATURES INCLUDE COARSE MESH AND CHEBYSHEV CONVERGENCE ACCELERATIONS, COARSE MESH SPATIAL AND ANGULAR ORGANIZATION TO PERMIT LARGER PROBLEMS, GENERAL ANISOTROPIC SCATTERING AND INHOMOGENEOUS SOURCE OPTION, INPUT SPECIFICATION OF TOP OR RIGHT BOUNDARY FLUXES, BUILT-IN DISCRETE ORDINATES CONSTANTS (S2, S4, ...S16), AND POSITIVE SPATIAL DIFFERENCE SCHEME.

8. RELATED AND AUXILIARY PROGRAMS - THE GENERAL GEOMETRY VERSION IS DESIGNED TO REPLACE ALL PRIOR VERSIONS OF 2DF (ACC ABSTRACT 173) AND TWOTRAN.

9. STATUS - ABSTRACT FIRST DISTRIBUTED SEPTEMBER 1969.
UNIVAC1108 X-Y VERSION SUBMITTED DECEMBER 1968, REPLACED BY UPDATED UNIVAC1108 VERSION JUNE 1970.
IBM360 X-Y VERSION SUBMITTED APRIL 1969.
UNIVAC1108 R-Z VERSION SUBMITTED OCTOBER 1969.
CDC6600 GENERAL GEOMETRY VERSION SUBMITTED JULY 1970.

10. REFERENCES - K. D. LATHROP, TWOTRAN, A FORTRAN PROGRAM FOR TWO-DIMENSIONAL TRANSPORT, GA-8747, JULY 22, 1968.
K. D. LATHROP, USERS GUIDE FOR THE TWOTRAN (X,Y) PROGRAM, LA-4058, FEBRUARY 14, 1969.
M. SHAPIRO, 2DF, REACTOR CODE ABSTRACT 173, ANL-7411, OCTOBER 1967.
K. D. LATHROP AND F. W. BRINKLEY, THEORY AND USE OF THE GENERAL GEOMETRY TWOTRAN PROGRAM, LA-4432, MAY 1970.
F. BRINKLEY, SUBROUTINE MANAGR, LASL NOTE, JULY 21, 1970.
RANDOM ECS I/O, LASL NOTE, JULY 10, 1969.

11. MACHINE REQUIREMENTS - FOR THE X-Y VERSION 4 OUTPUT UNITS (DISKS, DRUMS, OR TAPES) IN ADDITION TO INPUT AND OUTPUT UNITS. A VERSION IS AVAILABLE FOR THE CDC6600 WHICH USES EXTENDED CORE STORAGE INSTEAD OF THESE FOUR UNITS AND ALSO ALLOWS BIGGER PROBLEMS BY ELIMINATING MUCH OF THE FAST MEMORY REQUIREMENT. FOR THE GENERAL GEOMETRY VERSION 5 OUTPUT UNITS (DISKS, DRUMS, OR TAPES) ARE REQUIRED IN ADDITION TO 3 SYSTEM INPUT/OUTPUT UNITS. A CDC EXTENDED CORE STORAGE UNIT OR A LARGE BULK MEMORY IS ALSO REQUIRED. (DISKS, DRUMS, OR TAPES CAN BE SUBSTITUTED FOR THIS REQUIREMENT.)

12. PROGRAMMING LANGUAGE USED - FORTRAN IV WITH A SMALL AMOUNT OF MIXED INTEGER-FLOATING ARITHMETIC AND GENERALIZED SUBSCRIPTING IN THE X-Y AND GENERAL GEOMETRY VERSIONS AND MINOR USE OF 10H HOLLERITH FORMATS IN THE GENERAL GEOMETRY VERSION.

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II (UNIVAC1108), SCOPE 3.1 (CDC6600), AND OS/360 (IBM360).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE GENERAL GEOMETRY VERSION USES A LOCAL CROSS SECTION LIBRARY (SUBROUTINE LAXS), AN ALGORITHM TO REDUCE CORE SIZE (SUBROUTINE REDUCE), A SPECIAL DECIMAL DUMP FOR CDC MODE ERRORS, AND SPECIAL PLOTTING ROUTINES FOR CONTOUR AND PROJECTIVE FLUX DISPLAYS. THESE FEATURES MAY BE REMOVED SIMPLY.

15. NAME AND ESTABLISHMENT OF AUTHORS -

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (X-Y 1108-3909 CARDS, 360-3643 CARDS, R-Z 1108-3999 CARDS, GENERAL GEOMETRY 6600-8242 CARDS)
SAMPLE PROBLEMS (X-Y 1108-281 CARDS, 360-290 CARDS, R-Z 1108-107 CARDS, GENERAL GEOMETRY 6600-202 CARDS)
REFERENCE REPORTS, GA-8747, LA-4058, LA-4432, AND NOTES

17. CATEGORY - C
KEYWORDS - 2-DIMENSIONAL, TRANSPORT THEORY, SN METHOD, X-Y, CRITICALITY SEARCHES, FLUX DISTRIBUTION, ANISOTROPIC SCATTERING, 2DF CODES

REACTOR CODE ABSTRACT 359 10/69

1. NAME OR DESIGNATION OF PROGRAM - PUN1

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - PUN1 EVALUATES UNRESOLVED RADIATIVE CAPTURE INTEGRALS AND RELATED MULTIGROUP CROSS SECTIONS. THE UNRESOLVED DISTRIBUTIONS MAY HAVE VARIOUS ORBITAL ANGULAR MOMENTUM QUANTUM NUMBERS AND THE EFFECTS OF DOPPLER BROADENING AND SELF-SHIELDING ARE INCLUDED.

4. METHOD OF SOLUTION - THE PSI(THETA,X) FUNCTION IS EVALUATED ON THE INTERVAL (0,20/THETA) BY THE CFB ROUTINE DEVELOPED BY GELBARD (SEE REFERENCE 2). THE J(THETA,BETA) FUNCTION IS OBTAINED FROM THE PSI(THETA,X) FUNCTION BY TWO TRAPEZOIDAL INTEGRATIONS, FOLLOWED BY RICHARDSON EXTRAPOLATION PLUS A WING CORRECTION FOR X GREATER THAN 20/THETA. THE INTEGRAL INVOLVING CHI-SQUARE FLUCTUATION OF THE REDUCED NEUTRON WIDTH, GAMMA(NO), IS EVALUATED BY GAUSSIAN QUADRATURE. THE INTEGRATION OVER THE NEUTRON WEIGHTING SPECTRUM IS APPROXIMATED BY GAUSS QUADRATURE FOR THE 1/E CASE AND TRAPEZOIDAL INTEGRATION FOR OTHER WEIGHTING SPECTRA.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - NO MORE THAN 20 SPIN STATES PER CASE AND 54 ENERGY GROUPS ARE ALLOWED. AN ARBITRARY ENERGY WEIGHTING SPECTRUM OF UP TO 498 POINTS MAY BE USED, BUT ALL POINTS MUST BE POSITIVE (W(E) GREATER THAN 0).

6. TYPICAL RUNNING TIME - LESS THAN 1/4 SECOND PER SPIN STATE PER GROUP IS REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM -
   (A) ANGULAR MOMENTUM QUANTUM NUMBER L = 0, 1, 2, 3, 4 PERMITTED.
   (B) GAMMA(NO) PERMITTED TO FLUCTUATE WITH A CHI-SQUARE DISTRIBUTION OF UP TO 8 DEGREES OF FREEDOM FOR EACH SPIN STATE.
   (C) EVALUATION OF THE INTEGRAL CONTRIBUTION FROM LEVELS WITH GAMMA(NO) LESS THAN SOME FRACTION OF THE AVERAGE PERMITTED.
   (D) AN ARBITRARY GROSS ENERGY WEIGHTING SPECTRUM PERMITTED.
   (E) DANCOFF CORRECTION IS COMPUTED INTERNALLY FOR EITHER HEXAGONAL OR SQUARE LATTICE ARRAYS OF CYLINDRICAL CLAD FUEL RODS.

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED SEPTEMBER 1969.
   CDC6600 VERSION SUBMITTED APRIL 1969.

10. REFERENCES - N. M. STEEN, PUN-1 - A FORTRAN-IV PROGRAM FOR THE EVALUATION OF UNRESOLVED RESONANCE INTEGRALS AND RELATED MULTIGROUP CROSS SECTIONS (LWBR DEVELOPMENT PROGRAM), WAPD-TM-768,

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

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - A CALL TO SETECF IS COMPLETELY UNNECESSARY BUT ELIMINATES AN UNNECESSARY DIAGNOSTIC ON THE 1108.

15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
    SOURCE DECK (1175 CARDS)
    BINARY DECK (505 CARDS)
    SAMPLE PROBLEM (71 CARDS)
    LIBRARY (1512 CARDS)
    REFERENCE REPORT AND NOTE

17. CATEGORY - D
    KEYWORDS - 2-GROUP, ISOTOPES, PRODUCTION, DECAY, FISSION PRODUCTS, REACTION RATES

REACTOR CODE ABSTRACT 368 01/71

1. NAME OR DESIGNATION OF PROGRAM - FLANGE2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360, UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - FLANGE2 TAKES CROSS SECTIONS, ANGULAR DISTRIBUTION, RESONANCE PARAMETER, AND SCATTERING LAW DATA FROM ENDF/B DATA TAPES AND PREPARES THERMAL MULTIGROUP CROSS SECTIONS AND SCATTERING MATRICES.

4. METHOD OF SOLUTION - DIRECT INTEGRATION OF THE SCATTERING LAW IS USED TO OBTAIN LEGENDRE MOMENTS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
   MAXIMUM ENERGY GROUPS = 200
   MAXIMUM LEGENDRE ORDER = 5

6. TYPICAL RUNNING TIME - A CROSS SECTION CALCULATION ONLY REQUIRES APPROXIMATELY 1 MINUTE, WHILE A FULL SCATTERING MATRIX PROBLEM (L=5) TAKES ABOUT 10 MINUTES.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - FLANGE1 (ACC ABSTRACT 247)

9. STATUS - ABSTRACT FIRST DISTRIBUTED SEPTEMBER 1969.
   IBM360 VERSION SUBMITTED JANUARY 1969.
   UNIVAC1108 VERSION SUBMITTED MAY 1970.

10. REFERENCES - H. C. HONECK, Y. D. NALIBOFF, FLANGE-II, A CODE TO PROCESS THERMAL NEUTRON SCATTERING DATA FROM AN ENDF/B TAPE, PRELIMINARY REPORT (SECTIONS 1-5 ONLY), DECEMBER 1968, AND REVISIONS, SEPTEMBER 3, 1969.

11. MACHINE REQUIREMENTS - 32K WORDS

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360 (IBM360) AND EXEC II (UNIVAC1108).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - FLANGE2 REQUIRES FOR INPUT THE ENDF/B LIBRARY WHICH IS AVAILABLE FROM THE NATIONAL NEUTRON CROSS SECTION CENTER AT BROOKHAVEN NATIONAL LABORATORY.

15. NAME AND ESTABLISHMENT OF AUTHORS -
   IBM360, UNIVAC1108
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15. NAME AND ESTABLISHMENT OF AUTHOR(S) (CONTINUED)
UNIVAC1108

Y. D. NALIBOFF AND G. M. BORGONOVI
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P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (360-2839 CARDS, 1108-2788 CARDS)
SAMPLE PROBLEM (360-28 CARDS)
REFERENCE REPORT AND REVISIONS

17. CATEGORY - A
KEYWORDS - THERMAL MULTIGROUP CROSS SECTIONS, SCATTERING LAW, ANGULAR DISTRIBUTION, RESONANCE PARAMETERS, FLANGE CODES

REACTOR CODE ABSTRACT 369 10/69

1. NAME OR DESIGNATION OF PROGRAM - RELAP2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600, IBM7044, UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - RELAP2 CALCULATES FLOW, MASS INVENTORIES, TEMPERATURES, PRESSURES, REACTIVITIES, AND TRANSIENT POWER FOR THE PRIMARY SYSTEM OF A WATER REACTOR DURING A REACTIVITY OR A LOSS-OF-COOLANT ACCIDENT. ALTHOUGH RETAINING THE SIMPLIFIED GEOMETRY (THREE VOLUMES PLUS A CORE REGION) OF THE PREVIOUS RELAP PROGRAM, MANY IMPROVEMENTS AND EXTENSIONS HAVE BEEN MADE. THE GEOMETRY CAN BE MADE TO APPROXIMATE EITHER A PRESSURIZED OR A BOILING WATER REACTOR SYSTEM. THE CORE IS TREATED AS A TWO-POINT MODEL FOR POWER GENERATION, HEAT TRANSFER, AND REACTIVITY FEEDBACKS AND AS A ONE-POINT MODEL FOR THE REACTOR KINETICS, PRESSURE BALANCES, AND FLOW BALANCES. ALSO, RELAP2 CAN BE USED FOR REACTOR SYSTEM SAFETY STUDIES INCLUDING LARGE REACTIVITY EXCURSIONS AS WELL AS THE LOSS-OF-COOLANT AND PUMP-FAILURE ACCIDENTS.

4. METHOD OF SOLUTION - THE TABULAR VALUES OF PRESSURE ARE INVESTIGATED SUCCESSIVELY, STARTING AT THE PREVIOUS POINT IN THE STEAM TABLES, UNTIL BOTH THE KNOWN DENSITY AND ENTHALPY VALUES ARE BRACKETED. WITHIN THESE LIMITS THE CALCULATED PRESSURE IS CHANGED ITERATIVELY UNTIL THE DENSITY, CALCULATED FROM MULTIPOINT LINEAR INTERPOLATION FORMULAS, MATCHES THE KNOWN DENSITY WITHIN THE COMPUTER ACCURACY. A SUBROUTINE USING THE NEWTON-RAPHSON METHOD WAS WRITTEN WHICH CONVERGES THE FLOW EQUATIONS TO WITHIN COMPUTER ACCURACY LIMITS. NO INPUT INCREMENT IS REQUIRED AND NO FAILURES HAVE BEEN NOTED. THE MASS AND ENERGY DIFFERENTIAL EQUATIONS ARE SOLVED BY FORWARD FINITE DIFFERENCE TECHNIQUES.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE PRESSURIZED WATER REACTOR (PWR) SYSTEM IS DIVIDED INTO THREE BASIC VOLUMES - (A) THE COLD LEG WHICH INCLUDES THE FLUID FROM THE HEAT EXCHANGER TO THE REACTOR INLET, (B) THE HOT LEG FROM THE REACTOR OUTLET TO THE HEAT EXCHANGER, AND (C) A SYSTEM PRESSURIZER CONNECTED TO THE HOT VOLUME. EACH VOLUME IS DEFINED AS A SIMPLE CYLINDRICAL TANK WHERE RELATIVE ENTRANCE AND EXIT JUNCTIONS ARE SPECIFIED BY THE USER. SYSTEM BREAKS INVOLVING LEAKS ARE ALLOWED IN ANY OF THE THREE VOLUMES. THE TIME-STEPS MUST BE EMPIRICALLY DETERMINED BY THE USER.

6. TYPICAL RUNNING TIME - THE APPROXIMATE SPEED ON THE IBM7044 IS 200 TIME-STEPS PER MACHINE MINUTE. THE CDC6600 IS ABOUT TEN TIMES FASTER.

7. UNUSUAL FEATURES OF THE PROGRAM - RELAP2 RETAINS MOST OF THE CALCULATIONAL FEATURES OF ITS PREDECESSORS, BUT DIFFERS MAINLY IN THE REACTOR KINETICS, REACTOR CONTROL OPTIONS, TWO-PHASE SEPARATION MODELS, PRESSURE AND FLOW SEARCH TECHNIQUES, AND INPUT/OUTPUT FORM.

REACTOR CODE ABSTRACT 411 01/71

1. NAME OR DESIGNATION OF PROGRAM - M0661, M0657, M0626

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - M0661, M0657, AND M0626 PERFORM STATISTICAL ANALYSES OF DATA BASED ON A LEAST SQUARES POLYNOMIAL FIT.

4. METHOD OF SOLUTION - M0661 IS BASED ON THE CONSTANT RELATION N=B0 AND SIMPLY COMPUTES THE SAMPLE MEAN, STANDARD DEVIATION, AND VARIOUS OTHER STATISTICS FROM A GIVEN SET OF DATA. IT ALSO OBTAINS CONFIDENCE LIMITS FOR CERTAIN PARAMETERS BASED ON NORMAL DISTRIBUTION THEORY.

   M0657 USES THE STRAIGHT LINE N=B0+B1*X AS THE MODEL WHEN PAIRED DATA ARE GIVEN. ESTIMATES OF THE INTERCEPT, B0, AND THE SLOPE, B1, ARE OBTAINED AS ARE VARIOUS CONFIDENCE INTERVALS USEFUL IN A STATISTICAL ANALYSIS OF THE DATA.

   M0626 USES THE POLYNOMIAL OF THE R-TH DEGREE IN K VARIABLES AS ITS MODEL. IT PERFORMS A MULTIPLE LINEAR REGRESSION ANALYSIS OF THE DATA.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -

6. TYPICAL RUNNING TIME - LESS THAN 1 MINUTE IS REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM - THE DATA MAY BE TRANSFORMED BY ADDITION OF A CONSTANT, MULTIPLICATION BY A CONSTANT, RAISING TO A POWER, OR USE OF A LOGARITHMIC, EXPONENTIAL OR TRIGONOMETRIC TRANSFORMATION PRIOR TO THE ANALYSES.

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   CDC6600 VERSION SUBMITTED OCTOBER 1969.

10. REFERENCES - T. SHIMAMOTO, THREE COMPUTER PROGRAMS FOR POLYNOMIAL CURVE FITTING, WAPD-TM-914, AUGUST 1969.
    C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.

11. MACHINE REQUIREMENTS - CDC-280 PLOTTER FOR THE GRAPHICAL OUTPUT

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 3.1.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
    T. SHIMAMOTO

15. NAME AND ESTABLISHMENT OF AUTHOR(S) (CONTINUED)
BETTIS ATOMIC POWER LABORATORY
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WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (M0626 1514 CARDS, M0657 1300 CARDS, M0661 857 CARDS)
SAMPLE PROBLEMS (M0626 27 CARDS, M0657 16 CARDS, M0661 23 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS, WAPD-TM-668 AND WAPD-TM-914

17. CATEGORY - P
KEYWORDS - STATISTICS, LEAST SQUARES, DATA PROCESSING, GRAPHS

REACTOR CODE ABSTRACT 412 01/71

1. NAME OR DESIGNATION OF PROGRAM - MANE1

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE MANE1 PROGRAM FINDS THE MAGNETIC FLUX IN EACH BRANCH OF A MAGNETIC NETWORK CONSISTING OF A NUMBER OF BRANCHES OF IRON AND AIR. THE PROBLEM IS ONE STEP IN THE DESIGN OF SLOW SPEED RELUCTANCE MOTORS IN WHICH THE MAGNETIC FLUX IS DETERMINED IN ALL COMPONENT PARTS FOR A POSITION OF THE ROTOR WITH RESPECT TO THE STATOR.

4. METHOD OF SOLUTION - THE EQUATIONS TO BE SOLVED ARE OBTAINED BY APPLYING OHMS LAW AND KIRCHOFFS LAW TO THE MAGNETIC NETWORK. THE RESULTING NONLINEAR EQUATIONS ARE SOLVED BY THE NEWTON-RAPHSON METHOD OF ITERATION. THE LINEARIZED EQUATIONS ARE SOLVED BY USING KRONS FUNDAMENTAL FORMULA FOR MESH ANALYSIS OF AN ELECTRICAL CIRCUIT.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE NETWORK CONSISTS OF A NUMBER OF MAJOR RECTANGLES OR LOOPS CONNECTED TO EACH OTHER IN A CYCLIC FASHION. EACH LARGE RECTANGLE IS SUBDIVIDED INTO A NUMBER OF SMALLER RECTANGLES. EACH SIDE (NETWORK BRANCH) HAS A RELUCTANCE BUT ONLY ONE BRANCH OF EACH MAJOR RECTANGLE HAS A DRIVING FORCE CORRESPONDING TO THE APPLIED AMPERE TURNS IN THE STATOR. IF THERE ARE L LOOPS AND R RECTANGLES PER LOOP, THEN LR LINEAR EQUATIONS MUST BE SOLVED. A RULE OF THUMB LIMITATION IS LR LESS THAN OR EQUAL TO 90, ALTHOUGH THE PROGRAM IS ACTUALLY LIMITED ONLY BY THE AMOUNT OF CORE STORAGE AND THE ADEQUACY OF GAUSS ELIMINATION WITH PIVOTAL CONDENSATION.

6. TYPICAL RUNNING TIME - FOR VALUES OF LR = 24 THE SOLUTION IS OBTAINED IN 5 TO 10 SECONDS.

7. UNUSUAL FEATURES OF THE PROGRAM - GIVEN THE MAGNETIZATION CHARACTERISTICS OF THE VARIOUS STEELS (REPRESENTED BY POINTS ON A TABLE), MANE1 FINDS A POINT ON EACH CURVE AND CALCULATES THE EQUATION OF A TANGENT TO THE CURVE INSTEAD OF A STRAIGHT LINE FROM THE CALCULATED POINT TO THE ORIGIN. REGARDLESS OF THE COMPLICATION OF THE NETWORK, THE PROGRAM SOLVES THE PROBLEM IN LESS THAN 10 ITERATIONS. THE PROGRAM ALSO CALCULATES THE TOTAL MAGNETIC ENERGY IN THE ENTIRE NETWORK.

8. RELATED AND AUXILIARY PROGRAMS - MANE1 IS AN IMPROVED VERSION OF M0565.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6600 VERSION SUBMITTED OCTOBER 1969.

10. REFERENCES - D. A. GUERDAN, O. J. MARLOWE, MANE1 - A DIGITAL COMPUTER PROGRAM TO SOLVE A RECTANGULAR MAGNETIC NETWORK, WAPD-TM-790, SEPTEMBER 1969.

10. REFERENCES (CONTINUED)

D. A. GUERCAN, MO565, A PROGRAM TO SOLVE A RECTANGULAR MAGNETIC NETWORK FOR USE ON THE PHILCO-2000 COMPUTER, WAPD-TM-557, FEBRUARY 1966.

C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.

11. MACHINE REQUIREMENTS - FOR SMALL PROBLEMS, LR = 18, 35,000 (OCTAL) CORE LOCATIONS ARE REQUIRED. FOR LARGE PROBLEMS, LR = 80, 154,000 (OCTAL) CORE LOCATIONS ARE REQUIRED.

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 3.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - FORTRAN INTEGER AND REAL VARIABLES ARE EQUIVALENCED. IT IS ASSUMED THAT VARIABLES ARE ALLOCATED THE SAME NUMBER OF STORAGE BITS INDEPENDENTLY OF THIS TYPE DESIGNATION.

15. NAME AND ESTABLISHMENT OF AUTHORS -

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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (699 CARDS)
SAMPLE PROBLEM (14 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS, WAPD-TM-790 AND WAPD-TM-668

17. CATEGORY - I
KEYWORDS - DESIGN, MAGNETIC FLUX, MO565 CODES

REACTOR CODE ABSTRACT 413 01/71

1. NAME OR DESIGNATION OF PROGRAM - ALPHA-M

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - ALPHA-M IS USED FOR DETERMINING RADIOISOTOPES BY LEAST SQUARES RESOLUTION OF THE GAMMA RAY SPECTRA. IT CAN HANDLE A VERY LARGE LIBRARY OF GAMMA RAY SPECTRA AND TAKES INTO ACCOUNT CORRECTIONS SUCH AS BACKGROUND SUBTRACTION, COUNTING TIME, DECAY TIME, DEAD TIME, AUTOMATIC COMPENSATION FOR GAIN AND THRESHOLD SHIFTS, SIZE OF THE ALIQUOT, VOLUME REDUCTION PRIOR TO COUNTING, AND SO ON.

4. METHOD OF SOLUTION - ALPHA-M USES AN ITERATIVE LEAST SQUARES SCHEME FOR CLOSELY ESTIMATING THE WEIGHTING FACTORS. IN THE FIRST ITERATION, THE WEIGHTING FACTORS ARE ESTIMATED BY USING THE ACTUAL NUMBER OF COUNTS PER CHANNEL, LEADING TO A CALCULATED NUMBER OF COUNTS. IN THE SECOND, THE FACTORS ARE MORE CLOSELY ESTIMATED BY USING THE CALCULATED VALUES FROM THE FIRST ITERATION. FURTHER ITERATIONS LEAD TO THE DESIRED DEGREE OF ACCURACY.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   80 ISOTOPES IN THE LIBRARY
   256 CHANNELS IN THE ANALYZER

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - ALPHA-M IS AN IMPROVED VERSION OF PROGRAM ALPHA.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   IBM360 VERSION SUBMITTED OCTOBER 1969, SAMPLE PROBLEM EXECUTED BY ACC.

10. REFERENCES - E. SCHONFELD, ALPHA M - AN IMPROVED COMPUTER PROGRAM FOR DETERMINING RADIOISOTOPES BY LEAST-SQUARES RESOLUTION OF THE GAMMA-RAY SPECTRA, ORNL-3975, SEPTEMBER 1966.
    E. SCHONFELD, ALPHA - A COMPUTER PROGRAM FOR THE DETERMINATION OF RADIOISOTOPES BY LEAST-SQUARES RESOLUTION OF THE GAMMA-RAY SPECTRA, ORNL-3810, JULY 1965.

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 -

E. SCHONFELD
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OAK RIDGE, TENNESSEE 37830

16. MATERIAL AVAILABLE -
SOURCE DECK (692 CARDS)
SAMPLE PROBLEM (766 CARDS)
REFERENCE REPORT, ORNL-3975

17. CATEGORY - O
KEYWORDS - GAMMA RADIATION, SPECTRA, ISOTOPES, LEAST SQUARES, BACKGROUND, ALPHA CODES

REACTOR CODE ABSTRACT 414 01/71

1. NAME OR DESIGNATION OF PROGRAM - TAC3D

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - TAC3D IS DESIGNED TO TREAT TRANSIENT, THREE-DIMENSIONAL HEAT TRANSFER PROBLEMS. STEADY-STATE PROBLEMS ARE TREATED BY CONSIDERING THE PROBLEM TO BE A TRANSIENT, STARTING WITH AN ASSUMED TEMPERATURE DISTRIBUTION AND RUNNING UNTIL EQUILIBRIUM CONDITIONS ARE ESTABLISHED. GEOMETRICALLY, THE PROBLEM MAY BE DEFINED BY EITHER RECTANGULAR (X,Y,Z) OR CYLINDRICAL (R,Z,THETA) COORDINATES.

4. METHOD OF SOLUTION - THE HEAT CONDUCTION EQUATION IS REPLACED BY AN EQUIVALENT SET OF LINEAR FINITE DIFFERENCE EQUATIONS. THESE ARE SOLVED BY AN IMPLICIT ALTERNATING DIRECTION SCHEME WHICH REQUIRES A REGULAR GEOMETRY IN THAT THE POINTS AT WHICH TEMPERATURES ARE TO BE CALCULATED MUST BE IN REGULAR ROWS, COLUMNS, AND PLANES. AS A CONSEQUENCE, TAC3D IS PRIMARILY SUITED TO SOLVE PROBLEMS THAT ROUGHLY FIT AN ENVELOPE OF EITHER A RECTANGULAR PARALLELEPIPED OR AN INCOMPLETE RIGHT CIRCULAR CYLINDER.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
   (A) THE GRID PLANE SYSTEM MUST BE ORTHOGONAL IN THE RECTANGULAR OR CYLINDRICAL COORDINATE SYSTEM. THEREFORE, THE SIDES OF THE NODAL ELEMENTS MUST ALSO BE ORTHOGONAL. THE ENTIRE PROBLEM MUST BE BOUNDED BY 6 GRID PLANES ON ONE OF THE COORDINATE SYSTEMS. DIFFICULTIES IN TREATING IRREGULAR BOUNDARIES CAN BE OVERCOME TO SOME EXTENT THROUGH THE USE OF MATERIALS HAVING SPECIALLY CHOSEN PROPERTIES.
   (B) ALL RADIATION IS TREATED ONE-DIMENSIONALLY.
   (C) THERE ARE NO PROVISIONS FOR THERMAL EXPANSION OR CHANGE OF PHASE. SUCH SPECIAL HEAT TRANSFER SITUATIONS COULD BE INCLUDED BY EXTENSIONS OF THE EXISTING PROGRAMMING.

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM -
   (A) THE GEOMETRICAL INPUT IS SIMPLE.
   (B) THE INPUT OF THERMAL PARAMETERS IS BY FORTRAN V ARITHMETIC STATEMENT FUNCTIONS. MANY OF THE CALCULATION VARIABLES (TIME, LOCAL TEMPERATURE, LOCAL POSITION, ETC.) ARE AVAILABLE FOR USE IN THESE FUNCTIONS.
   (C) INTERNAL AND EXTERNAL FLOWING COOLANTS MAY BE USED.
   (D) THERE MAY BE INTERNAL AND EXTERNAL THERMAL RADIATION.
   (E) THERE IS A WIDE SELECTION OF OPTIONAL OUTPUT.
   (F) THERE IS A SPECIAL OPTION WHICH MAY BE USED FOR OBTAINING STEADY-STATE RESULTS EFFICIENTLY.

8. RELATED AND AUXILIARY PROGRAMS - TAC2D (ACC ABSTRACT 408), WHICH IS TWO-DIMENSIONAL AND HAS ALL THE FEATURES OF TAC3D, AND RAT3D, WHICH TAC3D REPLACED.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
UNIVAC1108 VERSION SUBMITTED OCTOBER 1969.

10. REFERENCES - J. F. PETERSEN, TAC3D, A GENERAL PURPOSE THREE-DIMENSIONAL HEAT TRANSFER COMPUTER CODE - USERS MANUAL, GA-9263, SEPTEMBER 1969.

S. S. CLARK, J. V. DEL BENE, AND J. F. PETERSEN, TAC3D, A GENERAL PURPOSE THREE-DIMENSIONAL HEAT TRANSFER COMPUTER CODE - MATHEMATICAL FORMULATIONS AND PROGRAMMERS GUIDE, GA-9264, SEPTEMBER 1969.

J. DOUGLAS, ALTERNATING DIRECTION METHODS FOR THREE SPACE VARIABLES, NUMERICAL MATHEMATICS, VOL. 4, PP. 41-63, 1962.

J. F. PETERSEN, TAC2D, A GENERAL PURPOSE TWO-DIMENSIONAL HEAT TRANSFER COMPUTER CODE - USERS MANUAL, GA-8868, SEPTEMBER 1969.

S. S. CLARK AND J. F. PETERSEN, TAC2D, A GENERAL PURPOSE TWO-DIMENSIONAL HEAT TRANSFER COMPUTER CODE - MATHEMATICAL FORMULATIONS AND PROGRAMMERS GUIDE, GA-9262, SEPTEMBER 1969.

11. MACHINE REQUIREMENTS - 65K MEMORY. A MAXIMUM OF FOUR AND A MINIMUM OF NO TAPES ARE REQUIRED, DEPENDING UPON THE CODE OPTIONS BEING USED.

12. PROGRAMMING LANGUAGE USED - FORTRAN V

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 (7992 CARDS)
SAMPLE PROBLEM (98 CARDS)
REFERENCE REPORTS, GA-9263 AND GA-9264

17. CATEGORY - H
KEYWORDS - TRANSIENTS, HEAT TRANSFER, 3-DIMENSIONAL, COOLANTS, X-Y-Z, R-Z-THETA, TAC2D CODES, RAT3D CODES

REACTOR CODE ABSTRACT 415 01/71

1. NAME OR DESIGNATION OF PROGRAM - CEXE/INCEXE

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - CEXE SOLVES THE THREE-DIMENSIONAL XYZ TIME-DEPENDENT XENON SPATIAL OSCILLATION PROBLEM USING A MODIFIED ONE ENERGY GROUP THEORY AND A NODAL REPRESENTATION.

4. METHOD OF SOLUTION - CEXE SOLVES FOR THE SPATIAL NEUTRON SOURCE DISTRIBUTION WITH COUPLED DOPPLER AND MODERATOR TEMPERATURE FEEDBACKS. THE TIME DEPENDENCE OF THE IODINE AND XENON CONCENTRATIONS ARE BASED ON THE ASSUMPTION OF CONSTANT POWER DURING EACH TIME-STEP INTERVAL.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMUM REACTOR CORE SIZE REPRESENTATION IS RESTRICTED TO A NODAL CONFIGURATION OF 19 X 19 X 10 IN THE X, Y, Z DIRECTIONS, RESPECTIVELY.

6. TYPICAL RUNNING TIME - A TYPICAL PROBLEM REQUIRES 4 SECONDS PER SOURCE ITERATION FOR A 1000 NODE CORE WITH NO FEEDBACKS AND 7.5 SECONDS PER ITERATION FOR A 2600 NODE CORE WITH BOTH DOPPLER AND MODERATOR TEMPERATURE FEEDBACK.

7. UNUSUAL FEATURES OF THE PROGRAM - CEXE PROVIDES A FULL THREE-DIMENSIONAL REPRESENTATION AND INCLUDES ALL MAJOR FEEDBACK MECHANISMS WHICH ARE IMPORTANT IN XENON OSCILLATION PROBLEMS. MAXIMUM FLEXIBILITY HAS BEEN PROVIDED AND THE CODE CAN BE USED FOR REACTORS WITH BIDIRECTIONAL COOLANT FLOW PATTERNS.

8. RELATED AND AUXILIARY PROGRAMS - INCEXE PREPARES TAPE INPUT REQUIRED BY CEXE.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
IBM360 VERSION SUBMITTED OCTOBER 1969, SAMPLE PROBLEM EXECUTED BY ACC.

10. REFERENCE - R. KERN, D. C. WRIGHT, W. C. COPPERSMITH, AND Z. R. ROSZTOCZY, OPTIMUM SPATIAL FLUX CONTROL PROJECT PHASE I CEXE, A THREE-DIMENSIONAL NODAL METHOD FOR ANALYSIS OF XENON OSCILLATIONS AND CONTROL, CENC-3932-1-349, MAY 1969.

11. MACHINE REQUIREMENTS - 256K BYTES, 6 IBM2314 DISK DRIVES, AND 3 IBM2400 TAPE DRIVES

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
AUTHOR

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (CEXE 1702 CARDS, INCEXE 198 CARDS)
SAMPLE PROBLEMS (CEXE 151 CARDS, INCEXE 26 CARDS)
REFERENCE REPORT

17. CATEGORY - F
KEYWORDS - 3-DIMENSIONAL, X-Y-Z, DOPPLER FEEDBACK, MODERATORS TEMPERATURE FEEDBACK, 1-GROUP, IODINE, XENON, CONTROL, HEAVY WATER REACTORS, INCEXE CODES

REACTOR CODE ABSTRACT 416 01/71

1. NAME OR DESIGNATION OF PROGRAM - PARTI

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6500

3. NATURE OF PHYSICAL PROBLEM SOLVED - PARTI IS A GROUP COLLAPSING CODE WHICH DETERMINES THE OPTIMUM DISCRETE REPRESENTATION OF A VARIABLE FOR SUBSEQUENT REPETITIVE CALCULATIONS.

4. METHOD OF SOLUTION - PARTI USES THE RESULTS FROM AN INITIAL DETAILED CALCULATION (SUCH AS A SLOWING DOWN CALCULATION FOR AN ENERGY-DEPENDENT FLUX APPLICATION) AND REDUCES THE CORRESPONDING SPACE AND/OR ENERGY STRUCTURE TO AN OPTIMAL FEW-INTERVAL AND/OR FEW-GROUP REPRESENTATION IN ACCORDANCE WITH THE MINIMIZATION OF AN OBJECTIVE RESPONSE FUNCTION. THIS MINIMIZATION IS ATTAINED EMPLOYING A METHOD OF STEEPEST DESCENT IN PIECE-WISE CONSTANT, NON-CONVEX, MULTI-DIMENSIONAL PHASE-SPACE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - PARTI WILL OPTIMIZE THE DISCRETE REPRESENTATION OF AN ARBITRARY FUNCTION OF ONE OR TWO INDEPENDENT VARIABLES. THE PRESENT PROGRAM STORAGE ALLOCATION PERMITS UP TO 80 NODES FOR EACH COORDINATE OF A DETAILED REPRESENTATION AND ALLOWS COLLAPSING UP TO 40 NODES FOR EACH COORDINATE OF THE REDUCED REPRESENTATION.

6. TYPICAL RUNNING TIME - RUNNING TIMES OF TYPICAL CASES HAVE SELDOM EXCEEDED 10 SECONDS.

7. UNUSUAL FEATURES OF THE PROGRAM - PARTI PERFORMS GROUP OR MESH COLLAPSING IN AN OPTIMAL SENSE.

8. RELATED AND AUXILIARY PROGRAMS - PARTI MAY BE CONSIDERED TO BE AN AUXILIARY PROGRAM TO ALMOST ANY NEUTRON DIFFUSION THEORY OR NEUTRON TRANSPORT PROGRAM.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6500 VERSION SUBMITTED NOVEMBER 1969.

10. REFERENCES - A. A. HARMS AND J. A. STODDARD, PARTI, A CODE FOR OPTIMAL PARTITIONING OF ONE AND TWO VARIABLE PIECEWISE-CONTINUOUS FUNCTIONS IN REACTOR CALCULATIONS, TRW 4332.2.69-410, 1969.
A. A. HARMS AND N. J. MCCORMICK, OPTIMAL NEUTRON GROUP STRUCTURE, TRANSACTIONS AMERICAN NUCLEAR SOCIETY, VOL. 11, P. 527, 1968.
A. A. HARMS AND A. L. BABB, OPTIMAL AND NON-OPTIMAL NEUTRON STRUCTURES, TRANSACTIONS AMERICAN NUCLEAR SOCIETY, VOL. 12, P. 143, 1969.
A. A. HARMS, VARIABLE PHASE-SPACE NEUTRON TRANSPORT ANALYSIS, UNIVERSITY OF WASHINGTON THESIS, 1969.

11. MACHINE REQUIREMENTS - 32K MEMORY

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 - A RANDOM NUMBER GENERATOR WITH THE FUNCTION NAME RANF(X) IS USED.

15. NAME AND ESTABLISHMENT OF AUTHORS -

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DEPARTMENT OF ENGINEERING PHYSICS
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REDONDO BEACH, CALIFORNIA 90278

16. MATERIAL AVAILABLE -
SOURCE DECK (6500-331 CARDS)
SAMPLE PROBLEM (6500-31 CARDS)
REFERENCE REPORT, TRW 4332.2.69-410

17. CATEGORY - B
KEYWORDS - GROUP CONSTANTS, FEW-GROUP CROSS SECTIONS, INPUT DATA

REACTOR CODE ABSTRACT 417 01/71

1. NAME OR DESIGNATION OF PROGRAM - ATHENA4

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360/75,91

3. NATURE OF PHYSICAL PROBLEM SOLVED - ATHENA4 COMPUTES FORM FACTORS FOR INELASTIC SCATTERING CALCULATIONS, USING SINGLE-PARTICLE WAVE FUNCTIONS THAT ARE EIGENSTATES OF MOTION IN EITHER A WOODS-SAXON POTENTIAL WELL OR A HARMONIC OSCILLATOR WELL. TWO-BODY FORCES OF GAUSS, COULOMB, YUKAWA, AND A SUM OF CUT-OFF YUKAWA RADIAL DEPENDENCES ARE AVAILABLE.

4. METHOD OF SOLUTION -

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   100 WAVE FUNCTION COMPONENTS
   10 INTERACTION COMPONENTS
   THE REDUCED MATRIX ELEMENTS OF THE ONE-BODY MULTIPOLE OPERATORS MAY BE INCLUDED AUTOMATICALLY IN THE WAVE FUNCTION COEFFICIENTS. NON-LOCAL EFFECTS ON THE WAVE FUNCTIONS MAY BE INCLUDED IN THE LOCAL ENERGY APPROXIMATION.

6. TYPICAL RUNNING TIME - ABOUT 2 MINUTES ARE REQUIRED FOR THE SAMPLE PROBLEM.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - ATHENA4 IS A REVISED AND UPDATED VERSION OF THE ORIGINAL ATHENA CODE. IT CAN PUNCH THE FORM FACTORS ON CARDS IN A FORMAT SUITABLE FOR INPUT TO THE DWBA CODE JULIE.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   IBM360 VERSION SUBMITTED NOVEMBER 1969, SAMPLE PROBLEM EXECUTED BY ACC.

10. REFERENCES - F. S. CHWIEROTH, J. I. DODSON, M. B. JOHNSON, L. W. OWEN, AND G. R. SATCHLER, FORM FACTORS FOR INELASTIC SCATTERING - THE CODE ATHENA IV, ORNL-TM-2703, SEPTEMBER 5, 1969, AND CORRIGENDA, APRIL 1970.

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
   F. S. CHWIEROTH, J. I. DODSON, M. B. JOHNSON,

15. NAME AND ESTABLISHMENT OF AUTHOR(S) (CONTINUED)

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (1926 CARDS)
SAMPLE PROBLEM (11 CARDS)
REFERENCE REPORT AND CORRIGENDA

17. CATEGORY - A
KEYWORDS - INELASTIC SCATTERING, POTENTIAL SCATTERING, NUCLEAR MODELS, ATHENA CODES, JULIE CODES

REACTOR CODE ABSTRACT 418 01/71

1. NAME OR DESIGNATION OF PROGRAM - CHAINS

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - GE625,635

3. NATURE OF PHYSICAL PROBLEM SOLVED - CHAINS COMPUTES THE ATOM DENSITY OF MEMBERS OF A SINGLE RADIOACTIVE DECAY CHAIN. THE LINEARITY OF THE BATEMAN EQUATIONS ALLOWS TRACING OF INTERCONNECTING CHAINS BY MANUALLY ACCUMULATING RESULTS FROM SEPARATE CALCULATIONS OF SINGLE CHAINS. RE-ENTRANT LOOPS CAN BE TREATED AS EXTENSIONS OF A SINGLE CHAIN. LOSSES FROM THE CHAIN ARE ALSO TALLIED.

4. METHOD OF SOLUTION - THE BATEMAN EQUATIONS ARE SOLVED ANALYTICALLY USING DOUBLE PRECISION ARITHMETIC. POLES ARE AVOIDED BY SMALL ALTERATIONS OF THE LOSS TERMS. MULTIGROUP FLUXES, CROSS SECTIONS, AND SELF-SHIELDING FACTORS ENTERED AS INPUT ARE USED TO COMPUTE THE EFFECTIVE SPECIFIC REACTION RATES. THE ATOM DENSITIES ARE COMPUTED AT ANY SPECIFIED TIMES.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   50 MEMBERS IN A CHAIN
   100 ENERGY GROUPS
   100 TIME VALUES

6. TYPICAL RUNNING TIME - LESS THAN 10 SECONDS ARE REQUIRED ON A GE625.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - CHAINS IS SIMILAR TO CRUNCH AND ISOCRUNCH (ACC ABSTRACT 180).

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   GE635 VERSION SUBMITTED DECEMBER 1969.

10. REFERENCE - W. B. HENDERSON, PROGRAM CHAINS (NMP-856), GEMP-490, MARCH 27, 1967.

11. MACHINE REQUIREMENTS - 16K MEMORY

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 -

15. NAME AND ESTABLISHMENT OF AUTHOR -

W. B. HENDERSON
WESTINGHOUSE PWR SYSTEMS DIVISION
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P. O. BOX 355
PITTSBURGH, PENNSYLVANIA 15230

16. MATERIAL AVAILABLE -
    SOURCE DECK (230 CARDS)
    BINARY DECK (73 CARDS)
    SAMPLE PROBLEM (55 CARDS)
    REFERENCE REPORT

17. CATEGORY - D
    KEYWORDS - ISOTOPES, PRODUCTION, DECAY, SPACE-INDEPENDENT, REACTION RATES, CRUNCH CODES, ISOCRUNCH CODES

REACTOR CODE ABSTRACT 419 01/71

1. NAME OR DESIGNATION OF PROGRAM - CASCADE/CLUSTER

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090,94

3. NATURE OF PHYSICAL PROBLEM SOLVED - CASCADE IS A SIMULATION OF THE COLLISION CASCADE RESULTING FROM THE DISPLACEMENT OF A PRIMARY KNOCK-ON ATOM BY AN ENERGETIC NEUTRON IN A BODY-CENTERED-CUBIC OR FACE-CENTERED-CUBIC CRYSTAL STRUCTURE. CLUSTER ANALYZES THE SPATIAL DISTRIBUTION OF THE RESULTING RADIATION DAMAGE IN TERMS OF VACANCY AND INTERSTITIAL CLUSTERS, TAKING INTO ACCOUNT THE EXACT CRYSTAL STRUCTURE.

4. METHOD OF SOLUTION - DISCRETE SIMULATION IS USED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROGRAM - THE BASIC ASSUMPTION IS THAT THE SOLE MODE OF ATOMIC INTERACTION IS THE BINARY ELASTIC COLLISION. FOR ALPHA-IRON AND COPPER THE VALID ENERGY RANGE IS BELOW 50 KEV. IN PRACTICE, MEMORY RESTRICTIONS LIMIT THE PRIMARY KNOCK-ON ATOM TO AN ENERGY RANGE LESS THAN 25 KEV.

6. TYPICAL RUNNING TIME - ABOUT 7.5 MINUTES ARE REQUIRED FOR A 20 KEV CASCADE.

7. UNUSUAL FEATURES OF THE PROGRAM - FAP ROUTINES ARE USED TO STORE AND RETRIEVE INFORMATION FROM MEMORY ABOUT LATTICE SITE OCCUPANCY. TO FULLY UTILIZE MEMORY, THIS INFORMATION IS PACKED.

8. RELATED AND AUXILIARY PROGRAMS - CASCADE IS AN EXTENSION OF EARLIER PROGRAMS FOR A SQUARE PLANAR LATTICE AND FOR A WURTZITE STRUCTURE.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
IBM7090 VERSION SUBMITTED NOVEMBER 1969.

10. REFERENCES - D. G. BESCO AND N. R. BAUMGARDT, CASCADE AND CLUSTER (COMPUTER PROGRAMS TO SIMULATE RADIATION DAMAGE PROCESSES IN METALS AND ANALYZE THE DISTRIBUTION OF DEFECTS), GEMP-356, APRIL 1965.
D. G. BESCO AND J. R. BEELER, JR., COMPUTER PROGRAMS DESCRIBING COLLISION CASCADES IN BINARY METALS, PART I - SQUARE PLANAR LATTICE, GEMP-192, FEBRUARY 1963, PART II - WURTZITE STRUCTURE, GEMP-200, FEBRUARY 1963, PART III - BODY-CENTERED-CUBIC AND FACE-CENTERED-CUBIC STRUCTURES, GEMP-243, AUGUST 1963.

11. MACHINE REQUIREMENTS - 32K MEMORY AND 6 TAPES

12. PROGRAMMING LANGUAGES USED - FORTRAN II (90 PERCENT)
FAP (10 PERCENT)

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

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATICN OR RESTRICTIONS - THE USE OF ASSEMBLY LANGUAGE FCR CRITICAL PARTS OF THE PROGRAM WILL REQUIRE SCME PROGRAMMING EFFCRT FOR CONVERSICN TO ANOTHER MACHINE.

15. NAME AND ESTABLISHMENT OF AUTHCRS -

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
    SCURCE CECK (7538 CARDS)
    BINARY CECK (1529 CARDS)
    SAMPLE PROBLEM (25 CARDS)
    REFERENCE REPORTS

17. CATEGORY - Q
    KEYWCRDS - SPECTRA, ELASTIC SCATTERING, CRYSTALS, RADIATION EFFECTS

REACTOR CODE ABSTRACT 420 01/71

1. NAME OR DESIGNATION OF PROGRAM - GROUSE

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - GE625,635

3. NATURE OF PHYSICAL PROBLEM SOLVED - GROUSE COMPUTES EFFECTIVE MULTIGROUP CROSS SECTIONS AS A FUNCTION OF POSITION IN THE CORE OF A REACTOR OR THE ABSORBER REGION OF A FUEL OR CONTROL ELEMENT. THE FLUX WEIGHTING USES SYNTHESIZED FLUXES GENERATED IN FINE ENERGY DETAIL FROM PARAMETRIC FLUX TRAVERSES SUPPLIED AS INPUT AS A FUNCTION OF THE CORE ABSORPTION AND SCATTER CROSS SECTIONS.

4. METHOD OF SOLUTION - CORE CROSS SECTIONS IN SUFFICIENTLY FINE ENERGY DETAIL TO PERFORM NUMERICAL DOPPLER-BROADENING ARE GENERATED FROM SINGLE-LEVEL BREIT-WIGNER PARAMETERS OVER THE SPECIFIED ENERGY RANGE. INPUT PARAMETRIC FLUX TRAVERSES ARE COMPUTED EXTERNALLY TO GROUSE BY ANY APPROPRIATE MONOENERGETIC TRANSPORT APPROXIMATION, WITH A TRIAL SOURCE IN THE CORE FOR WHICH A CORRECTION IN GROUSE IS SUBSEQUENTLY MADE. THE MODERATOR (REFLECTOR) AND CLAD (STRUCTURE) CROSS SECTIONS ARE SUPPLIED AS CONSTANT SCATTER AND ABSORPTION IN THE MONOENERGETIC FLUX TRAVERSE CALCULATIONS, BUT 1/V ABSORPTION IN THOSE REGIONS IS ALLOWED IN COMPUTING THE ENERGY DEPENDENCE OF THE MODERATOR FLUX. THE SPATIAL FLUX SHAPE AT EACH ENERGY IN THE FINE ENERGY LATTICE IS SYNTHESIZED BY INTERPOLATION IN THE PARAMETRIC VALUES USING THE VALUE OF THE CORE ABSORPTION AND SCATTER CROSS SECTIONS AT THOSE ENERGIES. THE GROUP REDUCTION USES THESE FLUXES TO GENERATE THE EFFECTIVE MULTIGROUP REACTION RATES IN SPECIFIED ZONES IN THE CORE. THE MULTIGROUP EFFECTIVE CROSS SECTIONS ARE COMPUTED TO REPRODUCE THE EFFECTIVE REACTION RATES BY AGAIN USING THE PARAMETRIC FLUX DATA.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   - 99 GROUPS
   - 40 CORE ZONES
   - 20 PARAMETRIC VALUES OF CORE ABSORPTION CROSS SECTION
   - 3 PARAMETRIC VALUES OF SCATTERING CROSS SECTION
   - 25 ISOTOPES
   - 2600 FINE ENERGY LATTICE POINTS

   THE CORE CAN CONTAIN A MIXTURE OF ISOTOPES IF THE ISOTOPIC COMPOSITION IS NOT A FUNCTION OF POSITION. IF ISOTOPIC COMPOSITION VARIES, EFFECTIVE CROSS REACTIONS FOR EACH ISOTOPE CAN BE CARRIED OUT ON ONE ISOTOPE AT A TIME, USING THE AVERAGE NON-RESONANT CROSS SECTION OF ALL OTHERS AS PART OF THE BACKGROUND CROSS SECTION.

6. TYPICAL RUNNING TIME - ABOUT 20 MINUTES ARE REQUIRED TO PROCESS 2600 ENERGIES AND 40 CORE ZONES TO OBTAIN MULTIGROUP CROSS SECTIONS INVOLVING 7 ISOTOPES. GENERATION OF DOPPLER-BROADENED ISOTOPIC LIBRARY TAPE IS PROPORTIONAL TO THE NUMBER OF ISOTOPES AND THE NUMBER OF ENERGIES FOR EACH ISOTOPE. ONE ISOTOPE AT 2600 ENERGIES CAN BE ADDED IN ABOUT 4 MINUTES.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - THE MONOENERGETIC PARAMETRIC FLUX DISTRIBUTIONS MUST BE CALCULATED AND PROCESSED EXTERNALLY TO GROUSE.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
GE635 VERSION SUBMITTED DECEMBER 1969.

10. REFERENCES - W. B. HENDERSON, DEVELOPMENT AND CORRELATION OF A METHOD FOR GENERATING SPACE-DEPENDENT MULTIGROUP EFFECTIVE CORE CROSS SECTIONS FOR 710 REACTOR ANALYSIS, GEMP-548, SEPTEMBER 25, 1967.
W. B. HENDERSON, USERS DATA FOR THE GROUSE CODE, GEMP NOTE, DECEMBER 2, 1962.

11. MACHINE REQUIREMENTS - 38K MEMORY, 6 TAPES OR DISC

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 - ENERGY LATTICES ARE ORDERED IN THE ANALOGUE OF A CARD SORTER. THE FLOATING NUMBERS ARE SORTED BIT BY BIT. THE SORTING ALGORITHMS (ORDER AND ORDER 1) ARE THEREFORE MACHINE-DEPENDENT AND ARE CODED FOR THE GE625,635 MACHINES.

15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (1814 CARDS)
BINARY DECK (563 CARDS)
SAMPLE PROBLEM (955 CARDS)
REFERENCE REPORT AND NOTE

17. CATEGORY - B
KEYWORDS - CROSS SECTIONS, BREIT-WIGNER FORMULA, DOPPLER BROADENING, FLUX SYNTHESIS, HETEROGENEOUS FLUX DISTRIBUTION, CELL CALCULATIONS, HOMOGENEOUS REACTORS, MULTIGROUP GROUP CONSTANTS

REACTOR CODE ABSTRACT 421 01/71

1. NAME OR DESIGNATION OF PROGRAM - MICHRD

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - GE625,635

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE VICKERS PYRAMID NUMBER AND THE MICROHARDNESS INCREMENT FRACTION ARE COMPUTED FROM FILAR MICROMETER EYEPIECE READINGS. THESE READINGS ARE MADE ON INDENTS, WHICH HAVE BEEN MADE BY AN INDENTER, IN THE MATERIAL BEING TESTED.

4. METHOD OF SOLUTION - THE VICKERS PYRAMID NUMBER IS A FUNCTION OF THE INCLUDED ANGLE BETWEEN TWO OPPOSITE FACES OF THE INDENTER, THE LOAD ON THE INDENTER, AND THE DIAGONAL OF THE INDENT. THE MICROHARDNESS INCREMENT FRACTION IS A RELATIONSHIP BETWEEN THE VICKERS PYRAMID NUMBERS OF TWO SPECIMENS, BOTH BEING TESTED AT THE SAME TEMPERATURE. IF READINGS FROM BOTH SPECIMENS ARE NOT AVAILABLE AT THE SAME TEMPERATURES, LINEAR INTERPOLATION ON THE GIVEN READINGS IS USED TO OBTAIN READINGS AT THE DESIRED TEMPERATURES.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   100 TEMPERATURES FOR EACH SPECIMEN
   1000 INDENTS, SUMMED OVER ALL TEMPERATURES, FOR EACH SPECIMEN
   4 READINGS PER INDENT

6. TYPICAL RUNNING TIME - LESS THAN 3 SECONDS PER CASE ARE REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - MICHRD IS WRITTEN TO USE A GE635 SUBROUTINE PLOP TO PLOT RESULTS WITH A CALCOMP PLOTTER. THIS IS A PROGRAM OPTION. THE PLOP SUBROUTINE FOR THE GE635 AND THE IBM7090 -94 IS MAINTAINED BY GENERAL ELECTRIC AIRCRAFT ENGINE GROUP.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   GE635 VERSION SUBMITTED DECEMBER 1969.

10. REFERENCE - N. R. BAUMGARDT AND J. L. KAMPHOUSE, HOT MICROHARDNESS PROGRAM, MICHRD, GEMP-725, DECEMBER 1, 1969.

11. MACHINE REQUIREMENTS - 27K MEMORY

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 -

15. NAME AND ESTABLISHMENT OF AUTHOR -
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AIRCRAFT ENGINE GROUP
CINCINNATI, OHIO 45215

16. MATERIAL AVAILABLE -
    SOURCE DECK (756 CARDS)
    BINARY DECK (216 CARDS)
    SAMPLE PROBLEM (195 CARDS)
    REFERENCE REPORT

17. CATEGORY - D
    KEYWORDS - MEASUREMENTS, RADIATION EFFECTS, TEMPERATURE, GRAPHS

REACTOR CODE ABSTRACT 422 01/71

1. NAME OR DESIGNATION OF PROGRAM - SPECTRA

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090-94

3. NATURE OF PHYSICAL PROBLEM SOLVED - SPECTRA COMPUTES THE NUMBER AND SPECTRUM OF PRIMARY KNOCK-ON ATOMS RESULTING FROM NEUTRON COLLISION SEQUENCES IN A MATERIAL UNDER IRRADIATION.

4. METHOD OF SOLUTION - MONTE CARLO TECHNIQUES ARE USED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROGRAM - ONLY ELASTIC SCATTER IS TREATED.

6. TYPICAL RUNNING TIME - THE SAMPLE PROBLEM, WHICH IS SMALL WITH 10,000 NEUTRON HISTORIES, REQUIRES 3.9 MINUTES.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - PROGRAM 20-4 AND THE SPECTRA INPUT PROGRAM ARE INCLUDED IN THE SPECTRA PACKAGE.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
IBM7090-94 VERSION SUBMITTED DECEMBER 1969.

10. REFERENCE - N. R. BAUMGARDT, SPECTRA PROGRAM USERS MANUAL, GEMP-728, DECEMBER 1969.

11. MACHINE REQUIREMENTS - 32K MEMORY

12. PROGRAMMING LANGUAGE USED - FAP

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

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - SINCE THIS PROGRAM IS WRITTEN IN ASSEMBLY LANGUAGE, IT WOULD REQUIRE CONSIDERABLE EFFORT TO CONVERT IT TO ANOTHER MACHINE.

15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (SPECTRA 3331 CARDS, 20-4 317 CARDS, 0659I 655 CARDS)
BINARY DECKS (SPECTRA 636 CARDS, 20-4 244 CARDS, 0659I 161 CARDS)
SAMPLE PROBLEMS (SPECTRA 63 CARDS, 20-4 355 CARDS, 0659I 151 CARDS)
REFERENCE REPORT

17. CATEGORY - Q
KEYWORDS - SPECTRA, RADIATION EFFECTS, ELASTIC SCATTERING, 20-4 CODES, 0659I CODES

REACTOR CODE ABSTRACT 423 01/71

1. NAME OR DESIGNATION OF PROGRAM - DOS

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - GE625,635

3. NATURE OF PHYSICAL PROBLEM SOLVED - DOS COMPUTES THE LOCAL NEUTRON FLUX PER REFERENCE REACTOR POWER LEVEL WHICH WILL PRODUCE THE MEASURED ACTIVITY OF A DOSIMETER. ALTERNATIVELY, IF THE FLUX VALUE IS SUPPLIED, THE CORRESPONDING DOSIMETER ACTIVITY WILL BE CALCULATED.

4. METHOD OF SOLUTION - THE TYPES OF DOSIMETERS TREATED ARE THOSE IN WHICH NEUTRON IRRADIATION OF THE PARENT ISOTOPE LEADS TO A GROUND STATE AND A METASTABLE STATE OF THE RADIOACTIVE DAUGHTER, OR TO A GROUND STATE ONLY. THE BATEMAN EQUATIONS ARE SOLVED DIRECTLY IF THE FLUX IS GIVEN AND ITERATIVELY IF THE ACTIVITY IS GIVEN.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMUM OF - 100 POWER CYCLES

6. TYPICAL RUNNING TIME - 0.25 SECONDS PER DOSIMETER ARE REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
GE635 VERSION SUBMITTED DECEMBER 1969.

10. REFERENCE - N. R. BAUMGARDT AND W. B. HENDERSON, PROGRAM DOS, NSP-874, A FORTRAN IV PROGRAM TO COMPUTE THE FLUX OR SPECIFIC ACTIVITY ASSOCIATED WITH A DOSIMETER, GEMP-729, DECEMBER 1969.

11. MACHINE REQUIREMENTS - 25K MEMORY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

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

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

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16. MATERIAL AVAILABLE -
    SOURCE DECK (456 CARDS)
    BINARY DECK (132 CARDS)
    SAMPLE PROBLEM (95 CARDS)
    REFERENCE REPORT

17. CATEGORY - D
    KEYWORDS - ISOTOPES, PRODUCTION, DECAY, SPACE-INDEPENDENT, DOSE RATES

REACTOR CODE ABSTRACT 424 01/71

1. NAME OR DESIGNATION OF PROGRAM - GLUB1

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - GLUB1 SOLVES THE WATERLOGGING TRANSIENT CAUSED BY THE EXISTENCE OF A FUEL ELEMENT CLADDING DEFECT DURING AN INCREASE IN POWER. THE POWER IS EITHER - CASE 1, PUT IN AS AN ARBITRARY FUNCTION OF TIME, OR CASE 2, ASSUMED TO BE A SERIES OF LINEAR RAMPS TO A NEW STEADY-STATE WITH STEEPER AND STEEPER SLOPES. ALL THE GEOMETRY AND METAL PROPERTIES ARE INPUT QUANTITIES. WATER PROPERTIES ARE OBTAINED FROM INTERNAL TABLES. CLAD AND FUEL THERMAL, STRESS, AND STRAIN COMPUTATIONS ARE PERFORMED. THE OUTPUT CONSISTS OF INTERNAL PRESSURE AND DEFECT FLOW. FOR CASE 1, THE CLAD STRESS VERSUS TIME IS OUTPUT, AND FOR CASE 2, THE NEW STEADY-STATE POWER WHICH JUST CAUSES CLAD FAILURE IS OUTPUT.

4. METHOD OF SOLUTION - THE TRANSIENT IS COMPUTED AS A SEQUENCE OF QUASI-STEADY STATES, WITH FUEL AND CLAD TEMPERATURES, STRESSES, AND STRAINS BEING COMPUTED AT ELEVEN DIFFERENT VALUES OF STEADY-STATE VOLUME AVERAGE ENTHALPY. LINEAR INTERPOLATION THEN GIVES THESE QUANTITIES AT ANY OTHER VOLUME AVERAGE ENTHALPY. THE COMPUTATIONS MUST BE DONE ITERATIVELY SINCE ALL METAL PROPERTIES CAN BE PUT IN AS TEMPERATURE DEPENDENT QUANTITIES. THEN STRAINS AND ENTHALPY ARE USED TO CALCULATE THE FUEL ELEMENT CORE, PLENUM, AND ANNULUS VOLUMES AND ENTHALPIES VERSUS TIME. FINALLY, THIS DATA IS USED TO COMPUTE INTERNAL PRESSURE, CLAD STRESSES, AND DEFECT FLOW. THE FUEL ELEMENT IS SPLIT INTO TWO PARTS, THE FUEL SECTION AND THE PLENUM. THE SPLIT OCCURS AXIALLY AT THE TOP OF THE FUEL. THE FUEL SECTION CONTAINS CONTROL VOLUMES FOR THE FUEL, CORE (IF ONE EXISTS), ANNULUS, AND CLAD. THE PLENUM SECTION HAS CONTROL VOLUMES FOR THE PLENUM AND CLAD. HEAT CONVECTION DUE TO WATER FLOWING BETWEEN SECTIONS AND OUT OF THE DEFECT IS SMALL AND ASSUMED NEGLIGIBLE. STRESS AND STRAIN CALCULATIONS ARE DONE ELASTICALLY WITH ALLOWANCE FOR INTERFERENCE BETWEEN THE FUEL AND CLAD.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - FOR EACH METAL PROPERTY, UP TO 10 VALUES VERSUS TEMPERATURE CAN BE PUT IN. IF POWER OR DEFECT ENTHALPY IS TO BE INPUT, UP TO 30 VALUES VERSUS TIME CAN BE GIVEN. FINALLY, IN ORDER TO OBTAIN ALLOWABLE POWER AS A FUNCTION OF DEFECT DIAMETER, UP TO 10 DIFFERENT DIAMETERS CAN BE PUT IN.

6. TYPICAL RUNNING TIME - 5 MINUTES ARE REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6600 VERSION SUBMITTED DECEMBER 1969.

10. REFERENCES - F. T. DUNCKHORST, L. L. LYNN, W. A. COFFMAN, AND J. E. MEYER, GLUP-1 - A FORTRAN IV DIGITAL COMPUTER PROGRAM FOR WATER-LOGGED FUEL ELEMENT ANALYSIS, WAPD-TM-569, NOVEMBER 1966.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.

11. MACHINE REQUIREMENTS - 60,000 OCTAL LOCATIONS

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 - THE OUTPUT CONVERSION AND PRINTING ROUTINES DO NOT PROPERLY PRINT THE (PHIO)MIN AND (PHIO)MAX SUBINTERVALS FOR THE TABLE OF JF, MEAN ENTHALPY OF THE FUEL.

15. NAME AND ESTABLISHMENT OF AUTHORS -
F. T. DUNCKHORST, L. L. LYNN, W. A. COFFMAN,
AND J. E. MEYER
BETTIS ATOMIC POWER LABORATORY
WESTINGHOUSE ELECTRIC CORPORATION
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (2638 CARDS)
SAMPLE PROBLEM (56 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS

17. CATEGORY - H
KEYWORDS - FUEL ELEMENTS, POWER TRANSIENTS, ELASTIC STRESSES, DEFORMATION, PRESSURE

REACTOR CODE ABSTRACT 425 01/71

1. NAME OR DESIGNATION OF PROGRAM - REDUX

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - REDUX IS USED FOR THE REDUCTION OF DATA FROM REACTOR FLUCTUATION EXPERIMENTS. FROM INPUT COUNT SAMPLES RECORDED BY ONE OR TWO COUNTING CHANNELS, THE PROGRAM COMPUTES UNBIASED SAMPLE ESTIMATES OF MOMENTS AND FUNCTIONS OF MOMENTS OF THE COUNT DISTRIBUTIONS, INCLUDING THE DISPERSION PARAMETER Y AND THE MODIFIED COEFFICIENT OF CORRELATION MCC. STANDARD ERROR ESTIMATES ARE COMPUTED FOR Y AND MCC.

4. METHOD OF SOLUTION - DIRECT COMPUTATION IS USED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   8 SAMPLES
   300 COUNT TRIALS PER SAMPLE

6. TYPICAL RUNNING TIME - 30 SECONDS ARE REQUIRED FOR THE LARGEST SAMPLES.

7. UNUSUAL FEATURES OF THE PROGRAM - BIAS AND ERROR ESTIMATES ARE USED FOR Y AND MCC.

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   CDC6600 VERSION SUBMITTED JANUARY 1970.

10. REFERENCES - D. R. HARRIS AND M. NATELSON, REDUX, A FORTRAN-IV PROGRAM FOR CALCULATION OF STATISTICAL PARAMETERS FROM COUNT SAMPLES MEASURED IN NEUTRON FLUCTUATION EXPERIMENTS. WAPD-TM-811, SEPTEMBER 1969.
   C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.

11. MACHINE REQUIREMENTS - 64,000 OCTAL LOCATIONS

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 AUTHORS -

D. R. HARRIS AND M. NATELSON
BETTIS ATOMIC POWER LABORATORY
WESTINGHOUSE ELECTRIC CORPORATION
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (273 CARDS)
SAMPLE PROBLEM (106 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS

17. CATEGORY - O
KEYWORDS - DATA PROCESSING, STATISTICS, REACTIVITY VARIATIONS

REACTOR CODE ABSTRACT 426 01/71

1. NAME OR DESIGNATION OF PROGRAM - PAX02

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - PAX02 IS USED TO GENERATE A HARMONY FILE CONTAINING FEW-GROUP MICROSCOPIC AND/OR MACROSCOPIC CROSS SECTIONS FOR PDQ7 PROBLEMS. INFINITE MEDIUM, TIME-DEPENDENT CALCULATIONS OF NEUTRON FLUX SPECTRA IN HOMOGENEOUS, HYDROGENOUS MEDIA ARE PERFORMED. HETEROGENEOUS RESONANCE INTEGRALS, THERMAL SHIELDING FACTORS, AND BLACKNESS PARAMETERS ARE CALCULATED. COEFFICIENTS FOR TEMPERATURE FEEDBACK CALCULATIONS ARE OBTAINED.

4. METHOD OF SOLUTION - INVENTORIES OF CELLS ARE INPUT IN THE FORM OF NUCLIDE CONCENTRATIONS OR IN THE FORM OF BASIC NUCLEAR MATERIALS AND THEIR GEOMETRIC ARRANGEMENTS AND SIZES. INDIVIDUAL MODULES OF THE PROGRAM CALCULATE SPECIFIC PROPERTIES OF THE CELL. THESE PROPERTIES, ALONG WITH OTHER RESULTS, ARE OUTPUT INTO A GENERAL FILE FOR FURTHER EDITING. ACCORDING TO PRESCRIBED PROCEDURES, THE HARMONY FILE REQUIRED BY PDQ7 IS GENERATED FROM THIS GENERAL FILE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   30 CELLS CALCULATED IN ONE PROBLEM
   30 NUCLIDES REPRESENTED IN THE FAST ENERGY SPECTRUM
   50 NUCLIDES USED IN OTHER PARTS OF THE PROGRAM
   4 FAST GROUPS WITH 1 THERMAL GROUP OR 3 FAST GROUPS WITH 2 OVERLAPPING THERMAL GROUPS
   3 PERTURBED PARAMETERS IN THE TEMPERATURE FEEDBACK CALCULATIONS
   UNLIMITED NUMBER OF DEPLETION TIMES

6. TYPICAL RUNNING TIME - ABOUT 15 SECONDS PER CELL PER DEPLETION STEP ARE REQUIRED EXCEPT FOR FINE DETAIL PLATE GEOMETRY PROBLEMS WHICH REQUIRE 90 SECONDS.

7. UNUSUAL FEATURES OF THE PROGRAM -
   (A) THE CELLS CAN BE DESCRIBED BY CONCENTRATIONS AND BASIC DIMENSIONS.
   (B) RESONANCE INTEGRALS CAN BE COMPUTED FROM INTERPOLATION FORMULAE FOR ROD GEOMETRY AND FROM FINE DETAIL MULTIGROUP CALCULATIONS FOR PLATE GEOMETRY.
   (C) FAST CROSS SECTIONS ARE OBTAINED FROM A 54-GROUP MILC LIBRARY.
   (D) THERMAL CROSS SECTIONS ARE OBTAINED FROM A KATE LIBRARY.
   (E) THERMAL SHIELDING FACTORS ARE OBTAINED WITH A DOUBLE-P APPROXIMATION FOR PLATES OR FROM A CORRELATION.

8. RELATED AND AUXILIARY PROGRAMS - THE FAST CROSS SECTION LIBRARY FILE INPUT TO PAX IS CREATED BY THE MILC6 PROGRAM. THE THERMAL CROSS SECTION LIBRARY AND THE THERMAL SCATTERING KERNELS ARE CREATED BY THE MO668 PROGRAM. HARMONY TABLESETS ARE INPUT TO PDQ7 (ACC ABSTRACT 275). THE FINK/PANTRY DATA STORAGE AND RETRIEVAL SYSTEM IS USED BY PAX02.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6600 VERSION SUBMITTED JANUARY 1970.

10. REFERENCES - H. BOHL, JR., R. J. BREEN, N. R. CANDELORE, O. J. MARLOWE, AND S. M. NIKOLICH, PAX, A CDC-6600 PROGRAM TO GENERATE HARMONY INPUT FOR PDQ, WAPD-TM-711(L), VOLS. I AND II, REVISED, NOVEMBER 1969.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.
H. BOHL, JR. AND C. J. MARLOWE, THE FINK/PANTRY DATA STORAGE AND RETRIEVAL SYSTEM FOR A COMPUTER, WAPD-TM-710, REVISED, JULY 1969.

11. MACHINE REQUIREMENTS - 64K CORE STORAGE AND AT LEAST ONE NON-SYSTEM DISK

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 3.1.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
H. BOHL, JR., R. J. BREEN, N. R. CANDELORE,
C. J. MARLOWE, AND S. M. NIKOLICH
BETTIS ATOMIC POWER LABORATORY
WESTINGHOUSE ELECTRIC CORPORATION
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (3 TAPES)
SOURCE DECKS (PAX02 18,528 CARDS, FINK/PANTRY 2268 CARDS, MILC6 806 CARDS, MO668 1202 CARDS)
SAMPLE PROBLEMS (PAX02 159 CARDS, MILC6 274 CARDS, MO668 69 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS

17. CATEGORY - B
KEYWORDS - CROSS SECTIONS, FEW-GROUP, THERMAL SPECTRA, RESONANCE INTEGRALS, FISSION PRODUCTS, HARMONY CODES, MILC6 CODES, MO668 CODES, PDQ7 CODES, FINK/PANTRY CODES

REACTOR CODE ABSTRACT 427 01/71

1. NAME OR DESIGNATION OF PROGRAM - FARED

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - FARED CONTAINS AN INTERNAL CROSS SECTION AVERAGING ROUTINE WHICH IS RESPONSIBLE FOR PREPARING BROAD GROUP CROSS SECTION SETS FOR VARIOUS MATERIAL REGIONS OF THE REACTOR. THE CROSS SECTION AVERAGING IS PERFORMED IN PROGRAM REGA, WHICH COMPUTES A B1 FLUX AND CURRENT IN UP TO 20 REACTOR BLOCK COMPOSITIONS FOR USE AS WEIGHTING FUNCTIONS IN THE CROSS SECTION COLLAPSING CALCULATION. A HOMOGENEOUS OR HETEROGENEOUS RESOLVED AND UNRESOLVED RESONANCE TREATMENT IS PROVIDED TO COMPUTE EFFECTIVE MICROGROUP RESONANCE CROSS SECTIONS FOR THE BLOCK MIXTURE OR UP TO 2 CELL TYPES PER BLOCK. REAL AND ADJOINT FLUX DISTRIBUTIONS ARE CALCULATED FOR ONE-DIMENSIONAL SLAB, CYLINDRICAL, OR SPHERICAL GEOMETRIES. THE REAL FLUXES ARE NORMALIZED TO YIELD DESIRED TOTAL REACTOR POWER. CRITICALITY SEARCHES MAY BE PERFORMED ON THE REACTOR DIMENSION, TRANSVERSE BUCKLING OR ZONE COMPOSITIONS. ENRICHMENT SEARCHES MAY BE PERFORMED TO YIELD DESIRED RATIOS OF MAXIMUM (OR AVERAGE) POWER DENSITIES IN SEVERAL ZONES. ZONEWISE DEPLETION IS CALCULATED EITHER FOR A GIVEN TIME PERIOD OR UNTIL SPECIFIED CRITICALITY, BURNUP OR NUCLIDE CONCENTRATIONS ARE SATISFIED. FLEXIBLE FUEL MANAGEMENT IS AVAILABLE PERMITTING SPECIFIED MATERIAL UNITS TO BE MOVED INTO, OUT OF OR SHUFFLED WITHIN THE REACTOR. A WIDE VARIETY OF EDITS MAY BE PERFORMED, INCLUDING PERTURBATION AND KINETIC PARAMETERS CALCULATIONS.

4. METHOD OF SOLUTION - THE PROCEDURE FOR COMPUTING BROAD GROUP MICROSCOPIC CROSS SECTIONS IS SIMILAR TO THAT USED IN THE GAM CODE (ACC ABSTRACT 33). FOR EACH HOMOGENEOUS BLOCK COMPOSITION, REGA PERFORMS A B1 COMPUTATION FOR THE FLUX AND CURRENT IN EACH MICROGROUP AND USES THESE AS WEIGHTING FUNCTIONS TO FORM MICROSCOPIC CROSS SECTION SETS FOR THE BROAD GROUP STRUCTURE SPECIFIED BY THE USER. FOR RESONANCE NUCLIDES, EFFECTIVE RESONANCE CAPTURE AND FISSION CROSS SECTION CONTRIBUTIONS ARE COMPUTED IN THE UNRESOLVED RESONANCE RANGE IN A MANNER SIMILAR TO THAT USED IN THE GANDY CODE (ACC ABSTRACT 341). A STATISTICAL AVERAGING OVER CHI-SQUARED DISTRIBUTION IS USED BUT INTERFERENCE SCATTERING IS IGNORED. AVERAGING OVER THE DISTRIBUTION IS PERFORMED USING THE QUADRATURE SCHEME INTRODUCED BY GREEBLER AND HUTCHINS. THE STRIP ROUTINE IN FARED IS RESPONSIBLE FOR THE CALCULATION OF EFFECTIVE RESONANCE CROSS SECTIONS IN THE RESOLVED RESONANCE RANGE. A MULTIGROUP COLLISION PROBABILITY PROCEDURE IS USED TO COMPUTE THE AVERAGE FLUX IN THE LUMP OR MIXTURE FOR EACH GROUP IN AN ULTRAFINE GROUP MESH WHICH SPANS THE ENTIRE RESOLVED RESONANCE RANGE. THE BOUNDARY FOR EACH ULTRAFINE GROUP IS COMPUTED BY STRIP AS THE CALCULATION PROCEEDS DOWN THE ENERGY RANGE. THE CALCULATION BEGINS AT THE HIGHEST ENERGY PEAK, ALL RESONANCE NUCLIDES CONSIDERED. NOTE THAT THIS PROCEDURE ACCOUNTS FOR OVERLAP AND INTERFERENCE EFFECTS. THE REAL AND ADJOINT DIFFUSION EQUATIONS ARE SOLVED BY A STANDARD FISSION SOURCE ITERATION PROCEDURE. A STARTING FISSION SOURCE DISTRIBU-

4. METHOD OF SOLUTION (CONTINUED)
TION IS PROVIDED BY THE USER. AFTER EACH ITERATION THE NEW FISSION SOURCE DISTRIBUTION IS NORMALIZED TO UNITY. THE DEPLETION EQUATIONS ARE SOLVED BY A SECOND-ORDER APPROXIMATION FOR EACH OF A SERIES OF SPECIFIED MINOR TIME-STEPS. THE POWER DISTRIBUTION IS ASSUMED CONSTANT DURING EACH MINOR TIME-STEP AND THE FLUXES ARE RENORMALIZED AT THE END OF EACH OF THESE INTERVALS. THE FLUX DISTRIBUTIONS ARE THEN RECALCULATED AT THE END OF THE SERIES OF MINOR TIME-STEPS. ALL CRITICALITY SEARCHES EMPLOY THE SAME GENERAL ALGORITHM FOR UNIFORMITY IN THE ITERATIVE PROCEDURE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
100 MICROGROUPS
30 MULTIGROUPS
15 DOWNSCATTERING GROUPS
150 MESH POINTS
50 UNIQUE NUCLIDES
20 ZONES
20 BLOCKS
100 ZONE MATERIALS
30 NUCLIDES PER BLOCK
20 ZONES PER BLOCK
2 CELLS PER BLOCK
15 NUCLIDES PER CELL COMPOSITION
30 NUCLIDES PER BLOCK COMPOSITION
7 RESONANCE NUCLIDES PER BLOCK

6. TYPICAL RUNNING TIME - THE EXECUTION TIME FOR THE SAMPLE PROBLEM GIVEN IN VOLUME 3 OF THE REFERENCE MANUAL WAS APPROXIMATELY 18 MINUTES. THIS PROBLEM INCLUDES THE CALCULATION OF THREE SETS OF 17-GROUP MICROSCOPIC CROSS SECTIONS (CORE, BLANKET, AND REFLECTOR), COMPOSITION SEARCHES, DEPLETION FOR SEVERAL TIME PERIODS, FUEL MANAGEMENT, AND EXTENSIVE EDIT CALCULATIONS.

7. UNUSUAL FEATURES OF THE PROGRAM - THE FARED CODE IS A TOTALLY INTEGRATED FAST REACTOR DESIGN PACKAGE. IT IS DESIGNED FOR FAST AND ACCURATE ONE-DIMENSIONAL PHYSICS DESIGN AND SURVEY STUDIES WITH A STRONG EMPHASIS ON INPUT SIMPLICITY. THE FOLLOWING ARE SOME OF THE SPECIAL FEATURES OF THE PROGRAM -
(A) THE USER IS PERMITTED TWO RESONANCE CELLS PER BLOCK TO TREAT, FOR EXAMPLE, A RESONANCE CONTROL MATERIAL AS WELL AS A FUEL LUMP CONTAINING RESONANCE NUCLIDES.
(B) THE MICROSCOPIC CROSS SECTIONS MAY BE RECALCULATED AT ANY TIME TO ACCOUNT FOR COMPOSITION CHANGES DURING DEPLETION.
(C) DEPLETION CHAINS ARE CONSTRUCTED BY THE USER AND NUCLIDE LISTS ARE EXPANDED TO INCLUDE ALL NUCLIDES IN THE CHAIN WHICH ARE PRODUCED DURING DEPLETION.
(D) THE BREEDING AND CONVERSION RATIOS ARE AUTOMATICALLY DEFINED BY SPECIFYING CERTAIN NUCLIDES AS FISSILE IN THE DEPLETION CHAIN.
(E) POWER DENSITY SEARCHES ARE AVAILABLE WHICH ADJUST MATERIAL COMPOSITIONS TO ACHIEVE THE DESIRED SYSTEM EIGENVALUE AND SPECIFIED RATIOS OF MAXIMUM (OR AVERAGE) POWER DENSI-

7. UNUSUAL FEATURES OF THE PROGRAM (CONTINUED)
TIES IN GIVEN SEARCH ZONES.
(F) NUCLIDE DEPLETION INCLUDES DESTRUCTION BY (N,ALPHA) AND (N,P) REACTIONS AND PRODUCTION BY (N,2N).
(G) THE LENGTH OF A DEPLETION SEQUENCE MAY BE DETERMINED BY THE MAXIMUM ALLOWABLE BURNUP OF CERTAIN MATERIAL UNITS, A MINIMUM ALLOWABLE CONCENTRATION OF CERTAIN NUCLIDES, OR A MINIMUM EIGENVALUE.
(H) A WIDE VARIETY OF EDITS ARE AVAILABLE INCLUDING EXTENSIVE PERTURBATION EDITS AND THE CALCULATION OF BETA-EFFECTIVE, PROMPT NEUTRON LIFETIME AND PROMPT AND DELAYED NEUTRON WORTHS.
(I) THE USE OF FREE FORMAT INPUT DIRECTIVES ALLOWS THE USER GREAT FLEXIBILITY IN SPECIFYING DATA AND CONSTRUCTING SEQUENCES OF CALCULATIONS AND EDITS.

8. RELATED AND AUXILIARY PROGRAMS - RETAP PREPARES A MICROGROUP LIBRARY FOR USE BY FARED. THE LIBRARY TAPE GENERATED BY THIS PROGRAM CONTAINS THE FOLLOWING SEVEN DATA FILES -
(1) MICROGROUP ENERGIES
(2) DEPLETION CHAIN DESCRIPTIONS
(3) RESOLVED RESONANCE DATA
(4) UNRESOLVED RESONANCE DATA
(5) SMOOTH CROSS SECTIONS AND MATRICES
(6) FISSION SPECTRA
(7) DEPLETION, FISSION PRODUCT AND DELAYED NEUTRON DATA.
AN UPDATE MAY BE PERFORMED ON AN EXISTING LIBRARY TO ADD OR CHANGE NUCLIDE DATA FOR ANY FILE.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC66CO VERSION SUBMITTED DECEMBER 1969.

10. REFERENCES - D. H. ROY, J. M. TILFORD, A. Z. LIVOLSI, P. N. COLPO, C. D. CARMICHAEL, AND J. A. JACOBSEN, FAST BREEDER STATIC PHYSICS METHODS DEVELOPMENT AND ANALYSIS PROJECT, VOLUME 1 - FARED, ONE-DIMENSIONAL FAST REACTOR PHYSICS DESIGN AND ANALYSIS CODE, VOLUME 2 - LISTING OF FARED AND RETAP PROGRAMS, VOLUME 3 - LISTING OF SAMPLE PROBLEM OUTPUT FROM RETAP AND FARED PROGRAMS, VOLUME 4 - PRODUCTION TESTS AND SENSITIVITY STUDIES USING THE FARED CODE, BAW-3867-9, OCTOBER 1969.

11. MACHINE REQUIREMENTS - 142,300 OCTAL CORE STORAGE AND 16 TAPE DRIVES OR DISTINCT DISC FILES PLUS I/O

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND COMPASS

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 3.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE TWO COMPASS MACHINE LANGUAGE ROUTINES ARE DESCRIBED IN THE USERS MANUAL AND ARE EASILY CONVERTIBLE TO ANOTHER MACHINE LANGUAGE. PROVISION IS MADE FOR VARYING THE NUMBER OF CHARACTERS PER

14. ANY OTHER INFORMATION OR RESTRICTIONS (CONTINUED)
WORD IN THE PROGRAM. FARED IS IN OVERLAY STRUCTURE WITH THREE PRIMARY OVERLAYS AND THIRTEEN SECONDARY OVERLAYS.

15. NAME AND ESTABLISHMENT OF AUTHORS -

D. H. ROY AND J. M. TILFORD
BABCOCK AND WILCOX
NUCLEAR POWER GENERATION DEPARTMENT
POWER GENERATION DIVISION
P. O. BOX 1260
LYNCHBURG, VIRGINIA 24505

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECKS (FARED 15,396 CARDS, RETAP 3920 CARDS)
SAMPLE PROBLEMS (FARED 81 CARDS, RETAP 16,473 CARDS)
REFERENCE REPORTS, VOLS. 1, 3, AND 4

17. CATEGORY - D
KEYWORDS - CROSS SECTIONS, AVERAGES, FLUX DISTRIBUTION, 1-DIMENSIONAL, SLABS, CYLINDERS, SPHERES, CRITICALITY SEARCHES, DEPLETION, PERTURBATION, REACTION RATES, RETAP CODES

REACTOR CODE ABSTRACT 428 01/71

1. NAME OR DESIGNATION OF PROGRAM - DOGGY

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - DOGGY CAN PERFORM MOST ROUTINE FORM SHEET CALCULATIONS. THE PROGRAM CAN HANDLE COMMON ARITHMETIC MANIPULATIONS ON COLUMNS OF INPUT SUCH AS ADDITION, SUBTRACTION, MULTIPLICATION, AND DIVISION. IT ALSO HAS PROVISIONS FOR THE USE OF SPECIAL FUNCTIONS SUCH AS LOGARITHMIC, TRIGONOMETRIC, ARC TRIGONOMETRIC, HYPERBOLIC, EXPONENTIAL, SQUARE ROOT, MAXIMA, MINIMA, AND RAISING A NUMBER TO ANY POWER. DOGGY ALSO CALCULATES WATER PROPERTIES SUCH AS THERMAL CONDUCTIVITY, VISCOSITY, AND PRANDTL NUMBER OVER A WIDE RANGE OF TEMPERATURE AND PRESSURE.

4. METHOD OF SOLUTION -

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   150 COLUMNS OF INPUT
   100 ROWS PER COLUMN

6. TYPICAL RUNNING TIME - RUNNING TIME RARELY EXCEEDS 30 SECONDS PER CASE.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - DOGGY IS AN UPDATED COMBINATION OF THE TWO FORMER KAPL FORM SHEET CALCULATION PROGRAMS, MPD AND DOGY.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   CDC6600 VERSION SUBMITTED JANUARY 1970.

10. REFERENCES - SALLY A. GAZDA, DOGGY, A FORTRAN-IV PROGRAM TO PERFORM MATHEMATICAL FORM SHEET CALCULATIONS, KAPL-M-6590, MARCH 15, 1967.
    H. H. RHODES, MPD1 - A PROGRAM TO PERFORM MATHEMATICAL FORM SHEET CALCULATIONS, JULY 8, 1964.
    J. R. SCHMID, DOGY - A PROGRAM TO PERFORM ROUTINE FORM SHEET CALCULATIONS, KAPL-M-EC-34, JUNE 19, 1963.
    R. J. CULLEN, 6600 CALCOMP PLOTTER ROUTINES, KAPL NOTE, JULY 25, 1966.

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASCENT

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 -

SALLY A. GAZDA
KNOLLS ATOMIC POWER LABORATORY
GENERAL ELECTRIC COMPANY
BOX 1072
SCHENECTADY, NEW YORK 12301

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (2618 CARDS)
SAMPLE PROBLEM (55 CARDS)
REFERENCE REPORT, KAPL-M-6590, AND NOTE

17. CATEGORY - P
KEYWORDS - DATA PROCESSING, NUMERICAL CALCULATIONS, GRAPHS, MPD CODES, DOGY CODES

REACTOR CODE ABSTRACT 429 01/71

1. NAME OR DESIGNATION OF PROGRAM - ASPIS

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - ASPIS COMPUTES THE ENERGY, DOSE AND ENERGY DEPOSITION BUILDUP FACTORS FOR MONOENERGETIC GAMMA RAYS FOR A PLANE ISOTROPIC, PLANE MONODIRECTIONAL, OR PLANE SLANT SOURCE, IN AN ARBITRARY LAMINAR ARRAY.

4. METHOD OF SOLUTION - ASPIS UTILIZES A SET OF LINEAR DIFFERENTIAL EQUATIONS, OBTAINED BY AN EXPANSION OF THE FLUX IN DOUBLE RANGE LEGENDRE POLYNOMIALS. THE CROSS SECTIONS ARE OBTAINED, BY INTERPOLATION, AT SPECIFIED ENERGY POINTS FROM A 25-POINT TABLE. THE GROUP CROSS SECTIONS AND THE SLOWING DOWN MATRIX ARE THEN OBTAINED FROM AN INFINITE MEDIUM SOLUTION IN EACH OF THE MATERIAL REGIONS. THE HIGHEST ENERGY GROUP IS INFINITELY THIN, AND THE UNSCATTERED SOURCE STRENGTH IS COMPUTED ANALYTICALLY THROUGHOUT THE SHIELD.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   - 12 REGIONS
   - 151 SPATIAL MESH POINTS PER REGION
   - 650 TOTAL SPATIAL MESH POINTS
   - 10 MATERIAL COMPOSITIONS
   - 10 MEV MAXIMUM ENERGY
   - 251 TOTAL ENERGY MESH POINTS
   - 13 ENERGY GROUPS
     - 6 TERMS IN THE DOUBLE RANGE LEGENDRE POLYNOMIAL EXPANSION OF THE FLUX
     - 6 TERMS IN THE LEGENDRE POLYNOMIAL EXPANSION OF THE SCATTERING KERNEL

6. TYPICAL RUNNING TIME - 2 MINUTES ARE REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM - ASPIS WILL COMPUTE THE ENERGY, DOSE, AND ENERGY DEPOSITION BUILDUP FACTORS FOR A PLANE SOURCE. FOR A MATERIAL COMPOSITION CONTAINING MORE THAN ONE ELEMENT THE ENERGY DEPOSITION BUILDUP FACTOR IN ANY ONE ELEMENT MAY BE COMPUTED.

8. RELATED AND AUXILIARY PROGRAMS - ASPIS IS A MODIFICATION OF LETO (ACC ABSTRACT 343) ADAPTED FOR MONOENERGETIC PLANE GAMMA RAY SOURCES.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   CDC6600 VERSION SUBMITTED FEBRUARY 1970.

10. REFERENCES - L. LOIS, ASPIS, A PROGRAM TO COMPUTE PLANE ISOTROPIC, MONODIRECTIONAL AND SLANT SOURCE BUILDUP FACTORS IN LAMINAR GEOMETRY, WAPD-TM-835, APRIL 1969.
    L. LOIS, A SOLUTION OF THE SLAB TRANSPORT PROBLEM, WAPD-TM-661, MARCH 1968.

10. REFERENCES (CONTINUED)

C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.

11. MACHINE REQUIREMENTS - 140K OCTAL MEMORY AND 3 TAPES FOR SCRATCH STORAGE

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASCENT

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 3.1.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE TERM OVERLAY ASSOCIATED WITH THE SCOPE 3.1 OPERATING SYSTEM DETERMINES THE AMOUNT OF PROGRAM TEXT IN CENTRAL MEMORY OF ANY MEMORY LOAD. THE ASPIS PROGRAM CONTAINS A MAIN OVERLAY AND 4 PRIMARY OVERLAYS. OVERALL PROGRAM CONTROL IS VESTED IN THE MAIN OVERLAY AND THE PRIMARY OVERLAYS ARE LOADED AND EXECUTED THROUGH THE USE OF THE OVERLAY ROUTINE.

15. NAME AND ESTABLISHMENT OF AUTHOR -

L. LOIS
BETTIS ATOMIC POWER LABORATORY
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WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (3706 CARDS)
SAMPLE PROBLEM (18 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS, WAPD-TM-835 AND WAPD-TM-668

17. CATEGORY - J
KEYWORDS - GAMMA RADIATION, SLABS, CROSS SECTIONS, SHIELDING, DOSE RATES, LETC CODES

REACTOR CODE ABSTRACT 430 01/71

1. NAME OR DESIGNATION OF PROGRAM - GAZE2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - GAZE2 IS A ONE-DIMENSIONAL, MULTIGROUP, NEUTRON DIFFUSION THEORY PROGRAM. IT INCLUDES ALL FOUR OF THE STANDARD ONE-DIMENSIONAL GEOMETRIES - SLAB, SPHERE, RADIAL CYLINDER, AND AXIAL CYLINDER, THE LAST OF WHICH IS IDENTICAL TO SLAB GEOMETRY EXCEPT WHEN THE ZOOM ALBEDO-TYPE TRANSVERSE BOUNDARY CONDITION IS USED.

4. METHOD OF SOLUTION - THIRD-ORDER FINITE DIFFERENCE EQUATIONS ARE USED AT EXTERIOR BOUNDARIES AND MATERIAL INTERFACES WITH FOURTH-ORDER DIFFERENCING AT INTERIOR POINTS. CRITICALITY SEARCHES ARE PERFORMED BY REPETITION OF PAIRS OF DIRECT AND ADJOINT FLUX CALCULATIONS, EACH PAIR FOLLOWED BY A FIRST-ORDER PERTURBATION TO DERIVE AN ESTIMATE OF THE PARAMETER VARIATION REQUIRED TO ACHIEVE THE DESIRED SYSTEM MULTIPLICATION FACTOR.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM - OPTIONAL TREATMENT OF THE TRANSVERSE BOUNDARY BY USE OF ENERGY- AND MATERIAL-DEPENDENT BUCKLINGS, LEAKAGE CROSS SECTIONS, OR ZOOM ALBEDO TYPE BOUNDARY CONDITIONS.

8. RELATED AND AUXILIARY PROGRAMS - ZOOM

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
UNIVAC1108 VERSION SUBMITTED JANUARY 1970.

10. REFERENCE - S. R. LENIHAN, GAZE-2, A ONE-DIMENSIONAL, MULTIGROUP, NEUTRON DIFFUSION THEORY CODE FOR THE IBM-7090, GA-3152, AUGUST 3, 1962.

11. MACHINE REQUIREMENTS -

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 (6121 CARDS)
    SAMPLE PROBLEM (205 CARDS)
    REFERENCE REPORT

17. CATEGORY - C
    KEYWORDS - 1-DIMENSIONAL, MULTIGROUP, DIFFUSION, SLABS, SPHERES, CYLINDERS, CRITICALITY SEARCHES, ZOOM CODES

REACTOR CODE ABSTRACT 431 02/71

1. NAME OR DESIGNATION OF PROGRAM - SUPERTOG

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - SUPERTOG ACCEPTS NUCLEAR DATA IN EITHER A POINT BY POINT OR PARAMETRIC REPRESENTATION AS SPECIFIED BY ENDF/B. THIS DATA IS AVERAGED OVER EACH SPECIFIED GROUP WIDTH. THE EXPLICIT ASSUMPTION IS MADE THAT THE FLUX PER UNIT LETHARGY IS CONSTANT OR THAT A SUITABLE WEIGHT FUNCTION WILL BE SUPPLIED BY THE USER. WHEN RESONANCE DATA IS AVAILABLE, RESOLVED AND UNRESOLVED RESONANCE CONTRIBUTIONS ARE CALCULATED AND USED AS SPECIFIED BY INPUT OPTIONS. FINE GROUP CONSTANTS SUCH AS ONE-DIMENSIONAL REACTION ARRAYS (ABSORPTION, FISSION, ETC.), PN ELASTIC SCATTERING MATRICES, AND INELASTIC AND (N,2N) SCATTERING MATRICES ARE GENERATED AND PLACED ON TAPES IN FORMATS SUITABLE FOR USE BY THE GAM1 (ACC ABSTRACT 33), GAM2, ANISN, OR DOT PROGRAMS.

4. METHOD OF SOLUTION - THE SINGLE-LEVEL BREIT-WIGNER FORMALISM IS USED FOR CALCULATION OF CROSS SECTIONS IN THE RESOLVED RESONANCE REGION. CROSS SECTIONS IN THE UNRESOLVED RESONANCE REGION ARE COMPUTED BY TAKING AVERAGES OVER SUITABLE PORTER-THOMAS DISTRIBUTIONS OF THE NEUTRON AND FISSION WIDTHS. SMOOTH CROSS SECTIONS ARE CALCULATED BY INTEGRATION OF POINT-CROSS-SECTION DATA GIVEN IN ENDF/B FILE 3. ELASTIC SCATTERING MATRICES ARE COMPUTED FROM LEGENDRE COEFFICIENTS OF THE SCATTERING ANGULAR DISTRIBUTION DATA. INELASTIC SCATTERING AND (N,2N) MATRICES ARE COMPUTED FROM EXCITATION FUNCTIONS FOR INDIVIDUAL LEVELS AND BY USING A NUCLEAR EVAPORATION MODEL ABOVE THE REGION OF RESOLVED LEVELS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   199 GROUPS
   4000 DATA POINTS FOR EACH REACTION TYPE
   30 LEGENDRE COEFFICIENTS

6. TYPICAL RUNNING TIME - ON THE IBM360/65 RUNNING TIME VARIES FROM 5 TO 25 MINUTES PER NUCLIDE AND IS A FUNCTION, PRIMARILY, OF THE NUMBER OF GROUPS, THE NUMBER OF RESOLVED RESONANCES, AND THE LENGTH OF THE ELASTIC SCATTERING MATRIX.

7. UNUSUAL FEATURES OF THE PROGRAM - SUPERTOG EMPLOYS A RATHER ELABORATE OVERLAY STRUCTURE AND EXTENSIVE USE OF EQUIVALENCE STATEMENTS TO MINIMIZE CORE REQUIREMENTS.

8. RELATED AND AUXILIARY PROGRAMS - THREE PROGRAMS ARE AVAILABLE WHICH EXTEND THE FLEXIBILITY OF SUPERTOG. THE SUPERTOG DATA RETRIEVAL PROGRAM RETRIEVES SUPERTOG OUTPUT FROM A CARD IMAGE TAPE WRITTEN IN THE ANISN FORMAT. THIS PROGRAM WILL RETRIEVE DATA FROM A MAXIMUM OF FOUR TAPES AND MERGE THIS DATA ONTO ONE TAPE. THE PROGRAM WILL THEN, BY INPUT OPTION, EDIT THE DATA, PUNCH CARDS IN EITHER THE ANISN OR DTF4 FORMAT, OR WRITE AN

8. RELATED AND AUXILIARY PROGRAMS (CONTINUED)
UNFORMATTED TAPE FOR USE BY ANISN. ANOTHER PROGRAM IS AVAILABLE WHICH WILL MERGE UP TO A MAXIMUM OF FOUR CARD IMAGE TAPES WRITTEN IN THE GAM2 UPDATE FORMAT ONTO A SINGLE TAPE. THIS PROGRAM CAN ALSO TAKE THE ONE-DIMENSIONAL ARRAYS FROM ONE TAPE AND THE TWO-DIMENSIONAL ARRAYS FROM ANOTHER TAPE AND MERGE THIS INFORMATION ONTO A SINGLE TAPE. THE SINGLE TAPE IS INPUT TO THE SUPERTOG VERSION OF THE GAM2 UPDATE PROGRAM. THE GAM2 UPDATE PROGRAM HAS BEEN MODIFIED TO ACCEPT OUTPUT FROM SUPERTOG ON EITHER PUNCHED CARDS OR MAGNETIC TAPE.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
IBM360 VERSION SUBMITTED FEBRUARY 1970, REPLACED BY VERSION 2 JANUARY 1971.

10. REFERENCES - R. Q. WRIGHT, J. L. LUCIUS, N. M. GREENE, AND C. W. CRAVEN, JR., SUPERTOG A PROGRAM TO GENERATE FINE GROUP CONSTANTS AND PN SCATTERING MATRICES FROM ENDF/B, ORNL-TM-2679, SEPTEMBER 1969, AND ADDENDUM, 1970.

11. MACHINE REQUIREMENTS - 65K MEMORY, 6 SCRATCH DISKS, AND 1 TAPE

12. PROGRAMMING LANGUAGE USED - FORTRAN IV (ASA STANDARD FORTRAN)

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - SUPERTOG IS PRESENTLY USED IN A TWO-LEVEL OVERLAY STRUCTURE CONSISTING OF SEVEN SEPARATE LINKS.

15. NAME AND ESTABLISHMENT OF AUTHORS -
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AND C. W. CRAVEN, JR.
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OAK RIDGE, TENNESSEE 37830

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (VERSION 1 - 1192 CARDS, VERSION 2 - 9883 CARDS)
SAMPLE PROBLEMS (VERSION 1 - 6 CARDS, VERSION 2 - 43 CARDS)
LIBRARIES (VERSION 1 - 1038 CARDS, VERSION 2 - 2231 CARDS)
REFERENCE REPORT AND ADDENDUM

17. CATEGORY - B
KEYWORDS - AVERAGES, RESOLVED REGION, UNRESOLVED REGION, GROUP CONSTANTS, ELASTIC, INELASTIC, SCATTERING, BREIT-WIGNER FORMULA, CROSS SECTIONS, LEGENDRE COEFFICIENTS, GAM1 CODES, GAM2 CODES, ANISN CODES, DOT CODES, DTF4 CODES, ETOG CODES

REACTOR CODE ABSTRACT 432 01/71

1. NAME OR DESIGNATION OF PROGRAM - COBRA2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108, IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - COBRA2 CALCULATES THE STEADY-STATE FLOW, ENTHALPY AND PRESSURE DROP IN THE SUBCHANNELS OF ROD BUNDLE NUCLEAR FUEL ELEMENTS DURING BOTH NON-BOILING AND BOILING CONDITIONS. THE PROGRAM USES A MATHEMATICAL MODEL THAT INCLUDES THE EFFECTS OF TURBULENT AND DIVERSION CROSSFLOW MIXING BETWEEN THE SUBCHANNELS.

4. METHOD OF SOLUTION - THE CONSERVATION EQUATIONS ARE SOLVED AS AN INITIAL-VALUE PROBLEM BY USING FINITE DIFFERENCES. THE SOLUTION MUST RELY UPON INPUT CORRELATIONS TO SPECIFY THE TURBULENT CROSSFLOW MIXING AND THE SUBCHANNEL PRESSURE GRADIENT.

5. RESTRICTIONS UPON THE COMPLEXITY OF THE PROBLEM - ADJUSTABLE DIMENSIONS ARE INCLUDED TO ALLOW THE USER TO EXPAND OR CONTRACT THE SIZE OF THE PROGRAM TO ACCOMODATE HIS COMPUTER STORAGE CAPABILITY. THE PROGRAM HAS PERFORMED SUCCESSFULLY WITH 36 SUBCHANNELS, 25 FUEL RODS AND 60 SUBCHANNEL CONNECTIONS.

6. TYPICAL RUNNING TIME - ABOUT 15 SECONDS PER CASE ARE REQUIRED FOR 9 SUBCHANNELS AND 144 AXIAL NODES.

7. UNUSUAL FEATURES OF THE PROGRAM - COBRA2 IS A RATHER GENERAL PROGRAM WHICH CAN BE USED FOR THERMALHYDRAULIC ANALYSIS OF MOST ROD BUNDLE FUEL ELEMENT CONFIGURATIONS. IT CAN CONSIDER DISTORTED BUNDLES BY USING VARIABLE SUBCHANNEL AREA AND GAP SPACING. IT CAN CONSIDER BOTH THE PRESSURE LOSS AND FORCED DIVERSION CROSSFLOW EFFECTS FROM SPACING DEVICES. IT CAN ALSO CONSIDER ARBITRARY AXIAL, RADIAL, AND CIRCUMFERENTIAL POWER DISTRIBUTION.

8. RELATED AND AUXILIARY PROGRAMS - COBRA2 IS AN IMPROVED VERSION OF THE ORIGINAL COBRA PROGRAM.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
UNIVAC1108 VERSION SUBMITTED FEBRUARY 1970.
IBM360 VERSION SUBMITTED FEBRUARY 1970, SAMPLE PROBLEM EXECUTED BY ACC.

10. REFERENCES - D. S. ROWE, COBRA-II, A DIGITAL COMPUTER PROGRAM FOR THERMAL-HYDRAULIC SUBCHANNEL ANALYSIS OF ROD BUNDLE NUCLEAR FUEL ELEMENTS, BNWL-1229, SEPTEMBER 1969.
D. S. ROWE, CROSSFLOW MIXING BETWEEN PARALLEL FLOW CHANNELS DURING BOILING, PART I, COBRA - COMPUTER PROGRAM FOR COOLANT BOILING IN ROD ARRAYS, BNWL-371, PT. 1, MARCH 1967.

11. MACHINE REQUIREMENTS - 64K MEMORY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II (UNIVAC1108) AND OS/360 (IBM360).

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 (1108-1635 CARDS, 360-1841 CARDS)
SAMPLE PROBLEMS (1108-94 CARDS, 360-98 CARDS)
REFERENCE REPORT, BNWL-1229

17 CATEGORY - H
KEYWORDS - THERMODYNAMICS, FLUID FLOW, FUEL ELEMENTS, LIQUIDS, VAPORS, ENTHALPY, PRESSURE, COBRA CODES

REACTOR CODE ABSTRACT 433 01/71

1. NAME OR DESIGNATION OF PROGRAM - CONTEMPT-CONPS

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE CONTEMPT-CONPS PROGRAM PREDICTS THE PRESSURE-TEMPERATURE RESPONSE OF A DRY WELL OR A DRY WELL WITH VENTS AND PRESSURE SUPPRESSION SYSTEM TO A LOSS-OF-COOLANT ACCIDENT. THE DRY WELL IS SEPARATED INTO A LIQUID REGION AND A VAPOR REGION. EACH REGION IS ASSUMED TO HAVE A UNIFORM TEMPERATURE BUT THE TEMPERATURES 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 MULTI-REGION HEAT CONDUCTION EQUATIONS. WATER AND ENERGY RATES FROM DISCHARGE OF COOLANT, BOILING OF RESIDUAL WATER BY REACTOR DECAY HEAT, SUPERHEATING OF STEAM PASSING THROUGH THE CORE, AND METAL-WATER REACTIONS ARE ASSUMED AVAILABLE FROM PREVIOUS CALCULATIONS AND ARE INPUT DATA TO THE PROGRAM. PROGRAM OUTPUT INCLUDES CONTAINMENT VOLUME PRESSURE AND TEMPERATURE, TEMPERATURES THROUGH THE BUILDING STRUCTURES, AND THE AMOUNT OF WATER, VAPOR, AND ENERGY IN THE CONTAINMENT VOLUMES. THE PRESSURE SUPPRESSION CALCULATIONS INCLUDE VENT CLEARING AND FLOW OF A TWO-COMPONENT TWO-PHASE WATER-AIR MIXTURE THROUGH THE VENTS, AND A MASS-ENERGY BALANCE IN THE WET WELL IN WHICH THE LIQUID AND VAPOR REGIONS ARE ASSUMED TO HAVE THE SAME TEMPERATURE. THERE ARE PROVISIONS FOR NORMAL BUILDING LEAKAGE, LEAKAGE FROM PENETRATIONS, A FAN COOLER SYSTEM, AND DRY AND WET WELL SPRAY SYSTEMS.

4. METHOD OF SOLUTION - THE INITIAL CONDITIONS OF THE CONTAINMENT ATMOSPHERE ARE CALCULATED FROM INPUT VALUES, AND THE INITIAL TEMPERATURE DISTRIBUTIONS 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. PRESSURE SUPPRESSION, SPRAY SYSTEM EFFECTS, AND FAN COOLER EFFECTS ARE CALCULATED USING CONDITIONS AT THE BEGINNING OF A TIME-STEP. LEAKAGE AND HEAT LOSSES OR GAINS, EXTRAPOLATED FROM THE LAST 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. THE PRESSURE SUPPRESSION ROUTINES USE THE CONDITIONS AT THE BEGINNING OF A TIME-STEP TO CALCULATE BOTH THE INITIAL EXPULSION OF WATER FROM THE VENTS AND THE FLOW THROUGH THE VENTS. FROM THE CALCULATED FLOW RATES, MASS AND ENERGY ARE REMOVED FROM THE DRY WELL AND ADDED TO THE WET WELL.

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 IBM360/75, APPROXIMATELY .021 SECOND PER TIME ADVANCEMENT WITH 90 MESH POINTS FOR HEAT STRUCTURES WITHOUT PRESSURE SUPPRESSION IS REQUIRED. THE PRESSURE SUPPRESSION TIMING IS NOT EASILY PREDICTED BUT TWO SAMPLE PROBLEMS RANGED FROM .3 TO 2 SECONDS PER TIME ADVANCEMENT.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - CONTEMPT, CONTEMPT-PS

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   IBM360 VERSION SUBMITTED FEBRUARY 1970.

10. REFERENCES - L. C. RICHARDSON, L. J. FINNEGAN, R. J. WAGNER, AND J. M. WAAGE, CONTEMPT, A COMPUTER PROGRAM FOR PREDICTING THE CONTAINMENT PRESSURE-TEMPERATURE RESPONSE TO A LOSS-OF-COOLANT ACCIDENT, IDO-17220, JUNE 1967.
    C. F. CARMICHAEL AND S. A. MARKO, CONTEMPT-PS - A DIGITAL COMPUTER CODE FOR PREDICTING THE PRESSURE-TEMPERATURE HISTORY WITHIN A PRESSURE-SUPPRESSION CONTAINMENT VESSEL IN RESPONSE TO A LOSS-OF-COOLANT ACCIDENT, IDO-17252, APRIL 1969.
    R. J. WAGNER, CONTEMPT MODIFICATIONS, PHILLIPS PETROLEUM MEMO WAG-24-68 AM, SEPTEMBER 23, 1968.
    R. J. WAGNER, CONTEMPT-CONPS MODIFICATIONS, WAG-19-69A-M, MAY 14, 1969.
    C. F. CARMICHAEL, MODIFICATIONS TO CONTEMPT/CONPS CODE, IDAHO NUCLEAR MEMO CARM-16-69, SEPTEMBER 17, 1969.
    L. C. RICHARDSON, DATE SUBROUTINE S0000010, PHILLIPS PETROLEUM NOTE, NOVEMBER 1968.
    R. J. WAGNER, CVI INPUT SUBROUTINE, IDAHO NUCLEAR NOTE, MAY 1969.
    R. L. MUELLER, 360/75 INTERVAL TIMER ROUTINE S00030, PHILLIPS PETROLEUM NOTE, OCTOBER 10, 1968.
    K. D. RICHERT, BUFIO, A SUBROUTINE TO PERMIT FORTRAN ACCESS TO IOOP, PHILLIPS PETROLEUM NOTE, FEBRUARY 1966.
    A. J. SMITH, CALCOMP PLOTTER SUBROUTINES, PHILLIPS PETROLEUM DESCRIPTION, FEBRUARY 1969.
    CONTEMPT-CONPS TRANSMITTAL INFORMATION, IDAHO NUCLEAR NOTE, OCTOBER 30, 1969.

11. MACHINE REQUIREMENTS - 250K BYTES AND A CALCOMP PLOTTER USED FOR PLOTTING

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND BAL

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360 MVT.

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 (6001 CARDS)
    SAMPLE PROBLEM (64 CARDS)
    JCL AND SAMPLE PROBLEM FOR LOAD MODULE (132 CARDS)
    CALCOMP WRITER ROUTINE (386 CARDS)
    LOAD MODULE CONTEMPT (FILE 1)
    REFERENCE REPORTS, NOTES, AND DESCRIPTION

17. CATEGORY - G
    KEYWORDS - ACCIDENTS, TEMPERATURE DISTRIBUTION, PRESSURE DISTRIBUTION, CONTAINMENT, WATER, THERMODYNAMICS, LEAKAGE, HEAT TRANSFER, CONTEMPT CODES

REACTOR CODE ABSTRACT 434 01/71

1. NAME OR DESIGNATION OF PROGRAM - HEATMESH

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - HEATMESH IS USED TO GENERATE GEOMETRICAL DATA REQUIRED FOR STUDIES OF HEAT TRANSFER IN AXISYMMETRIC STRUCTURES REPRESENTED AS SURFACES OF REVOLUTION. THE PROGRAM CONSISTS OF TWO DISTINCT PHASES. THE FIRST SUBDIVIDES THE GIVEN PARTS INTO A NODAL NETWORK AND EVALUATES THE GEOMETRICAL PROPERTIES OF THE NODES. THE SECOND DETERMINES ADJACENT NODES AND EDITS GEOMETRICAL DATA FOR THE THERMAL MODEL.

4. METHOD OF SOLUTION - THE STRUCTURE TO BE STUDIED, REPRESENTED AS A BODY OF REVOLUTION, IS DIVIDED INTO PARTS HAVING COMMON MATERIAL PROPERTIES AND REPRESENTED AS BODIES OF REVOLUTION. EACH PART IS THEN DESCRIBED AS FOUR SURFACES OF REVOLUTION SUBDIVIDED INTO NODES WHICH FORM A MESH. DATA FOR EACH PART ARE COLLECTED, I.E. VOLUME, AREA, AND PART NUMBER OF EACH NODE, AND NODE SURFACES ON THE PART BOUNDARY AND INSIDE THE PART BOUNDARY. THE DISTANCE BETWEEN THE CENTER AND THE MIDPOINT OF EACH SURFACE OF THE NODE IS TABULATED ALSO.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
   NUMBER OF SUBDIVISIONS BETWEEN 1 AND 50 FOR SIDES 1 AND 3
   NUMBER OF SUBDIVISIONS BETWEEN 1 AND 12 FOR SIDES 2 AND 4

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM - THE INPUT DATA ARE CONSTRUCTED IN THE SAME MANNER AS USED IN THE APT PROGRAM. DATA GENERATED BY APT CAN BE USED IN HEATMESH WITH ONLY MINOR CHANGES.

8. RELATED AND AUXILIARY PROGRAMS - THE DATA GENERATED BY HEATMESH ARE USED IN THE DIFFERENCE EQUATIONS OF THE HEATFLOW PROGRAM WHICH SOLVES TRANSIENT HEAT TRANSFER PROBLEMS.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   CDC6600 VERSION SUBMITTED JANUARY 1970.

10. REFERENCES - V. K. GABRIELSON, HEATMESH, A COMPUTER CODE FOR GENERATING GEOMETRICAL DATA REQUIRED FOR STUDIES OF HEAT TRANSFER IN AXISYMMETRIC STRUCTURES, SCL-DR-67-30, SEPTEMBER 1967, AND ERRATA, 1970.

11. MACHINE REQUIREMENTS -

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 - ALL SUBROUTINES CALLED WHICH ARE NOT INCLUDED REFER TO SANDIA (LIVERMORE) PLOT PROGRAMS AND SHOULD BE EASILY CONVERTED TO OTHER SYSTEMS.

15. NAME AND ESTABLISHMENT OF AUTHOR -

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16. MATERIAL AVAILABLE -
SOURCE DECK (1033 CARDS)
SAMPLE PROBLEM (45 CARDS)
REFERENCE REPORT AND ERRATA

17. CATEGORY - N
KEYWORDS - HEAT TRANSFER, GEOMETRIES, INPUT DATA, STRUCTURAL ANALYSIS

REACTOR CODE ABSTRACT 435 01/71

1. NAME OR DESIGNATION OF PROGRAM - BURST1

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360/75

3. NATURE OF PHYSICAL PROBLEM SOLVED - BURST1 PROVIDES FOR THE STUDY OF PRESSURIZED LIQUID IN A CYLINDRICAL SYSTEM IMMEDIATELY AFTER A RUPTURE OCCURS AT ONE OR BOTH ENDS OF THE SYSTEM. PRESSURE, MASS VELOCITY, AND ENTHALPY ARE CALCULATED AT EQUALLY-SPACED MESH POINTS ALONG THE LENGTH OF THE SYSTEM. THIS IS A ONE-DIMENSIONAL REPRESENTATION, ASSUMING UNIFORM CONDITIONS THROUGHOUT ANY GIVEN CROSS-SECTIONAL AREA. FORCES ON DESIGNATED SECTIONS OF THE SYSTEM CAN ALSO BE COMPUTED, WITH PROVISION FOR DIRECTIONAL CHANGES.

4. METHOD OF SOLUTION - PARTIAL TIME DERIVATIVES FOR MASS VELOCITY, PRESSURE, AND ENTHALPY, OBTAINED FROM THE THREE CONSERVATION LAWS FOR MASS, MOMENTUM, AND ENERGY, ARE APPROXIMATED WITH DIFFERENCE EQUATIONS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   50 TOTAL CYLINDERS AND ADAPTERS
   2000 SPACE STEPS IN THE WHOLE SYSTEM
   5000 TIME-STEPS
   ONLY SUBCOOLED, FRICTIONLESS, ADIABATIC CASES CAN BE CONSIDERED.

6. TYPICAL RUNNING TIME - WITH A LARGE PROBLEM ON THE IBM7044, THE PRODUCT OF THE NUMBER OF SPACE STEPS AND THE NUMBER OF TIME-STEPS RUN IN ONE MINUTE IS ABOUT 25,000.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - THE BLOT DECK IS USED TO PLOT DATA STORED BY BURST1.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   IBM360 VERSION SUBMITTED FEBRUARY 1970, SAMPLE PROBLEM EXECUTED BY ACC.

10. REFERENCE - RONALD P. ROSE, GEORGE H. HANSON, AND GERALD A. JAYNE, HYDRODYNAMICS DESCRIBING ACOUSTIC PHENOMENA DURING REACTOR COOLANT SYSTEM BLOWDOWN, ICO-17254, JULY 1967.

11. MACHINE REQUIREMENTS - 32K MEMORY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

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

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

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16. MATERIAL AVAILABLE -

SOURCE DECKS (BURST1 360-612 CARDS, BLOT 360-240 CARDS)
SAMPLE PROBLEMS (BURST1 360-33 CARDS, BLOT 360-29 CARDS)
REFERENCE REPORT

17. CATEGORY - G

KEYWORDS - CYLINDERS, PRESSURE DISTRIBUTION, 1-DIMENSIONAL, ENTHALPY, BLOWDOWN, HYDRODYNAMICS, COOLANTS, BLOT CODES

REACTOR CODE ABSTRACT 436 01/71

1. NAME OR DESIGNATION OF PROGRAM - ETOM1

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - ETOM1 PROCESSES BASIC NUCLEAR INFORMATION GIVEN IN THE ENDF/B FORMAT AND PRODUCES DATA DECKS FOR USE IN GENERATION OF MUFT4 AND MUFT5 LIBRARIES.

4. METHOD OF SOLUTION - BASICALLY, ETOM1 PERFORMS A STRAIGHTFORWARD GROUP AVERAGING OF THE VARIOUS CROSS SECTIONS USING THE INPUT WEIGHT AND THE MICROSCOPIC DATA FROM THE ENDF/B DATA FILES. THE BREIT-WIGNER SINGLE-LEVEL FORMULA IS ASSUMED TO YIELD RESONANCE CROSS SECTIONS WITH REASONABLE ACCURACY.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   99 MULTIGROUPS
   250 RESOLVED RESONANCES

6. TYPICAL RUNNING TIME - EXECUTION TIMES FOR A STANDARD 54-GROUP MUFT PROBLEM USING ENDF/B DATA WHICH CONTAIN EXPLICIT RESONANCE PARAMETERS VARY FROM 1 TO 2 MINUTES. WHEN THE ENDF/B DATA FOR THE ISOTOPE DOES NOT CONTAIN EXPLICIT RESONANCE PARAMETERS, EXECUTION TIMES VARY FROM .5 TO 1 MINUTE.

7. UNUSUAL FEATURES OF THE PROGRAM - THE GENERAL PURPOSE RETRIEVAL SUBROUTINES ENABLE TWO TABULAR FUNCTIONS TO BE ACCURATELY COMBINED (TO WITHIN SOME INPUT EPSILON) AND THEY ALSO PROVIDE FOR EXACT (ACCORDING TO THE SPECIFIED INTERPOLATION SCHEMES) INTEGRATION OF THE FUNCTIONS. ANY DESIRED WEIGHTING SCHEME MAY BE USED. ETOM1 WILL ACCEPT ENDF/B DATA FROM A BINARY TAPE OR FROM A BCD CARD IMAGE TAPE OR JUST FROM BCD CARDS. ALL DATA IS ASSUMED TO BE IN THE STANDARD ENDF/B ARRANGEMENT. MANY OF THE SUBROUTINES USED BY THE PROGRAM MAY BE USEFUL IN OTHER (PRESENT AND FUTURE) CODES CONNECTED WITH THE ENDF/B SYSTEM.

8. RELATED AND AUXILIARY PROGRAMS - STORAGE AND RETRIEVAL OF THE ENDF/B MICROSCOPIC DATA AND COMBINATION OF TWO OR MORE TABULAR ARRAYS INTO ONE TABULAR ARRAY WITH ONE INTERPOLATION SCHEME IS ACCOMPLISHED IN LARGE PART BY THE GENERAL PURPOSE RETRIEVAL SUBROUTINES (ACC ABSTRACT 384).

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   CDC6600 VERSION SUBMITTED FEBRUARY 1970.

10. REFERENCES - R. A. DANNELS AND D. E. KUSNER, ETOM-1 - A FORTRAN PROGRAM TO PROCESS DATA FROM THE ENDF/B FILE TO THE MUFT FORMAT, WCAP-3688-1, MAY 1968.
    H. C. HONECK, ENDF/B - SPECIFICATIONS FOR AN EVALUATED NUCLEAR DATA FILE FOR REACTOR APPLICATIONS, BNL-50066, MAY 1966, REVISED JULY 1967.
    H. BOHL, JR., E. M. GELBARD, AND G. H. RYAN, MUFT-4

10. REFERENCES (CONTINUED)
FAST NEUTRON SPECTRUM CODE FOR THE IBM-704, WAPD-TM-72, 1957.
H. BOHL, JR. AND A. P. HEMPHILL, MUFT-5, A FAST NEUTRON SPECTRUM PROGRAM FOR THE PHILCO-2000, WAPD-TM-218, FEBRUARY 1961.
J. FELBERBAUM, ED., DESCRIPTION OF THE ENDF/B PROCESSING CODES CHECKER, CRECT, DAMMET, AND RETRIEVAL SUBROUTINES, BNL-13582 (ENDF-110), SEPTEMBER 1967, REVISED APRIL 1969.

11. MACHINE REQUIREMENTS - 43,000 (DECIMAL) LOCATIONS WITH 2 TAPES (SCRATCH TAPE AND LIBRARY TAPE)

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 3.1.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - ETOM1 IS SEGMENTED INTO 6 BASIC OVERLAYS.

15. NAME AND ESTABLISHMENT OF AUTHORS -
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (6600-7094 CARDS)
SAMPLE PROBLEM (6600-5 CARDS)
REFERENCE REPORT, WCAP-3688-1

17. CATEGORY - B
KEYWORDS - CROSS SECTIONS, BREIT-WIGNER FORMULA, GROUP CONSTANTS, LIBRARIES, AVERAGES, MUFT CODES

REACTOR CODE ABSTRACT 437 01/71

1. NAME OR DESIGNATION OF PROGRAM - ETOG1

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - ETOG1 PROCESSES BASIC NUCLEAR INFORMATION GIVEN IN THE ENDF/B FORMAT AND PRODUCES DATA DECKS FOR USE IN GENERATION OF MUFT4, MUFT5, GAM1, GAM2, AND ANISN LIBRARIES.

4. METHOD OF SOLUTION - ETOG1 PERFORMS A STRAIGHTFORWARD GROUP AVERAGE OF CROSS SECTIONS USING AN INPUT WEIGHTING FUNCTION AND THE MICROSCOPIC DATA FROM THE ENDF/B FILES. THE BREIT-WIGNER SINGLE-LEVEL FORMULA IS ASSUMED TO YIELD RESOLVED RESONANCE CROSS SECTIONS WITH REASONABLE ACCURACY. TREATMENT OF UNRESOLVED CROSS SECTIONS IS SIMILAR TO THAT OF MC**2 (ACC ABSTRACT 355).

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   99 MULTIGROUPS
   250 RESOLVED RESONANCES

6. TYPICAL RUNNING TIME -

	ENDF DATA HAS RESONANCE PARAMETERS	ENDF DATA HAS NO RESONANCE PARAMETERS
MUFT, 54 GROUP	1 MINUTE	.5 MINUTE
GAM1, 68 GROUP	7 MINUTES	4 MINUTES
GAM2, 99 GROUP	15 MINUTES	8 MINUTES
ANISN, 99 GROUP	15 MINUTES	8 MINUTES

7. UNUSUAL FEATURES OF THE PROGRAM - IN THE ESTABLISHMENT OF TRANSFER MATRICES, ANALYTICAL RESULTS ARE EMPLOYED WHERE POSSIBLE IN THE NORMALIZATION OF OUTPUT DATA. FOR EXAMPLE, IN THE GENERATION OF THE ELASTIC SCATTERING MATRIX IN THE ENERGY RANGE IN WHICH THE SCATTER IS ISOTROPIC IN THE CENTER-OF-MASS SYSTEM, RESULTS OF THE NUMERICAL INTEGRATION PERFORMED ARE NORMALIZED TO VALUES DERIVED ANALYTICALLY.

8. RELATED AND AUXILIARY PROGRAMS - ETOG1 SUPERSEDES ETOM1 (ACC ABSTRACT 436) IN THAT (A) MORE ACCURATE DATA DECKS ARE PRODUCED FOR USE IN CREATING MUFT LIBRARIES, AND (B) ADDITIONAL DATA DECKS MAY BE GENERATED FOR USE IN CREATING GAM AND ANISN LIBRARIES. IN THE PROGRAM USE IS MADE OF THE GENERAL PURPOSE PROCESSOR AND RETRIEVAL SUBROUTINES WRITTEN AT BROOKHAVEN FOR HANDLING ENDF/B DATA FILES (ACC ABSTRACT 384).

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   CDC6600 VERSION SUBMITTED JANUARY 1970, REPLACED BY LATER VERSION MAY 1970.

10. REFERENCES - D. E. KUSNER AND S. KELLMAN, ETOG-1, A FORTRAN IV PROGRAM TO PROCESS DATA FROM THE ENDF/B FILE TO THE MUFT, GAM AND ANISN FORMATS, WCAP-3845-1 (ENDF-114), DECEMBER 1969, AND ERRATA.

H. C. HONECK, ENDF/B SPECIFICATIONS FOR AN EVALUATED NUCLEAR DATA FILE FOR REACTOR APPLICATIONS, BNL-50066, MAY 1966, REVISED JULY 1967.

H. BOHL, JR., E. M. GELBARD, AND G. H. RYAN, MUFT-4, FAST NEUTRON SPECTRUM CODE FOR THE IBM-704, WAPD-TM-72, 1957.

H. BOHL, JR. AND A. P. HEMPHILL, MUFT-5 - A FAST NEUTRON SPECTRUM PROGRAM FOR THE PHILCO-2000, WAPD-TM-218, FEBRUARY 1961.

G. D. JOANOU AND J. S. DUDEK, GAM-I, A CONSISTENT P1 MULTIGROUP CODE FOR THE CALCULATION OF FAST NEUTRON SPECTRA AND MULTIGROUP CONSTANTS, GA-1850, MAY 1961.

G. D. JOANOU AND J. S. DUDEK, GAM-II, A B3 CODE FOR CALCULATION OF SLOWING DOWN SPECTRUM AND ASSOCIATED MULTIGROUP CONSTANTS, GA-4265, 1963.

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

D. M. OSHEA, B. J. TOPPEL AND A. L. RAGO, MC**2 - A CODE TO CALCULATE MULTIGROUP CROSS SECTIONS, ANL-7318, JUNE 1967.

J. FELBERBAUM, ED., DESCRIPTION OF THE ENDF/B PROCESSING CODES CHECKER, CROCT, CAMMET, AND RETRIEVAL SUBROUTINES, BNL-13582 (ENDF-110), SEPTEMBER 1967, REVISED APRIL 1969.

11. MACHINE REQUIREMENTS - 55,000 DECIMAL LOCATIONS AND 4 SCRATCH TAPES IN ADDITION TO THE LIBRARY TAPE AND STANDARD INPUT, OUTPUT, AND PUNCH UNITS

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 3.1.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE PROGRAM CONSISTS OF 8 OVERLAYS.

15. NAME AND ESTABLISHMENT OF AUTHORS -

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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (12,329 CARDS)
SAMPLE PROBLEM (3 CARDS)
LIBRARY (1044 CARDS)
REFERENCE REPORT, WCAP-3845-1, AND ERRATA

17. CATEGORY - B
KEYWORDS - CROSS SECTIONS, BREIT-WIGNER FORMULA, GROUP CONSTANTS,

17. KEYWORDS (CONTINUED)
AVERAGES, LIBRARIES, ETOM CODES, GAM CODES, ANISN CODES, MUFT CODES

REACTOR CODE ABSTRACT 438 01/71

1. NAME OR DESIGNATION OF PROGRAM - BUG2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE TWO-DIMENSIONAL MULTIGROUP NEUTRON DIFFUSION THEORY EQUATIONS FOR X-Y OR R-Z GEOMETRY ARE SOLVED TO OBTAIN THE MULTIPLICATION FACTOR AND THE SPATIAL FLUX AND POWER DISTRIBUTIONS. COMPLETE REACTOR LIFE HISTORIES WITH PARTIAL REFUELING AT A NUMBER OF RELOAD POINTS CAN BE CALCULATED. THE DEPLETION SCHEME OF ALL BURNABLE NUCLIDES IS SPECIFIED BY THE USER AT EXECUTION TIME. A REGIONWIDE DEPLETION SCHEME IS USED. CONCENTRATION DEPENDENT SELF-SHIELDING FACTORS MAY BE APPLIED TO ANY NUCLIDE.

4. METHOD OF SOLUTION - OVER AN ARBITRARY UNIFORM MESH GRID, THE SYSTEM OF DIFFERENCE EQUATIONS FOR THE STEADY-STATE SOLUTION IS SOLVED BY AN EXTENSION OF THE POWER METHOD. A FORM OF SUCCESSIVE LINE OVERRELAXATION IS USED IN EACH GROUP. COARSE MESH REBALANCING IS USED TO IMPROVE PREASYMPTOTIC CONVERGENCE BEHAVIOR AND AN EXPONENTIAL EXTRAPOLATION METHOD IS USED TO IMPROVE ASYMPTOTIC BEHAVIOR. FOR DEPLETION, THE CODE DECIDES INTERNALLY (FOR EACH TIME-STEP AND EVERY NUCLIDE) WHETHER TO USE EITHER A DIFFERENCE APPROXIMATION OR AN ANALYTICAL SOLUTION. A FULL TRIANGULAR DEPLETION MATRIX IS ALLOWED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   - 10 GROUPS
   - 20,000 SPACE MESH POINTS
   - 255 REGIONS
   - 300 SUBREGIONS
   - 40 TOTAL NUCLIDES
   - 25 HEAVY NUCLIDES
   - 25 FISSION PRODUCTS AND OTHER BURNABLE NUCLIDES

6. TYPICAL RUNNING TIME - AN X-Y PROBLEM WITH 4 GROUPS AND 7400 MESH POINTS REQUIRES 65 MINUTES TO DO 8 TIME-STEPS. AN X-Y PROBLEM WITH 9 GROUPS AND 7400 MESH POINTS REQUIRES 155 MINUTES FOR 8 TIME-STEPS.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - GAMBLE5 (ACC ABSTRACT 222) IS USED FOR THE STEADY-STATE SOLUTIONS.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   UNIVAC1108 VERSION SUBMITTED MARCH 1970.

10. REFERENCE - J. P. DORSEY, R. FROEHLICH, AND F. TODT, BUG-2/BUGTRI, TWO DIMENSIONAL MULTIGROUP BURNUP CODES FOR RECTANGULAR AND HEXAGONAL GEOMETRY, GA-8272, AUGUST 22, 1969.

11. MACHINE REQUIREMENTS - 65,536 WORDS OF CORE STORAGE, 2 MILLION WORDS OF DRUM STORAGE ON ONE DATA CHANNEL, 3 TAPE UNITS, AND A PERIPHERAL PRINTER

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASSEMBLY LANGUAGE

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC-II, GAX23.

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 (21,874 CARDS)
    SAMPLE PROBLEM (155 CARDS)
    REFERENCE REPORT

17. CATEGORY - D
    KEYWORDS - 2-DIMENSIONAL, MULTIGROUP, DIFFUSION, DEPLETION, X-Y, R-Z, FLUX DISTRIBUTION, POWER DISTRIBUTION, GAMBLE5 CODES

REACTOR CODE ABSTRACT 439 01/71

1. NAME OR DESIGNATION OF PROGRAM - BUGTRI

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE TWO-DIMENSIONAL MULTIGROUP NEUTRON DIFFUSION THEORY EQUATIONS FOR TRIANGULAR GEOMETRY ARE SOLVED TO OBTAIN THE MULTIPLICATION FACTOR AND THE SPATIAL FLUX AND POWER DISTRIBUTIONS. COMPLETE REACTOR LIFE HISTORIES WITH PARTIAL REFUELING AT A NUMBER OF RELOAD TIME POINTS CAN BE CALCULATED. THE DEPLETION SCHEME OF ALL BURNABLE NUCLIDES IS SPECIFIED BY THE USER AT EXECUTION TIME. A REGIONWISE DEPLETION SCHEME IS USED. CONCENTRATION DEPENDENT SELF-SHIELDING FACTORS MAY BE APPLIED TO ANY NUCLIDE.

4. METHOD OF SOLUTION - OVER A UNIFORM TRIANGULAR MESH GRID, THE SYSTEM OF DIFFERENCE EQUATIONS FOR THE STEADY-STATE SOLUTION IS SOLVED BY AN EXTENSION OF THE POWER METHOD. A FORM OF SUCCESSIVE LINE OVERRELAXATION IS USED IN EACH GROUP. COARSE MESH REBALANCING IS USED TO IMPROVE PREASYMPTOTIC CONVERGENCE BEHAVIOR AND AN EXPONENTIAL EXTRAPOLATION METHOD IS USED TO IMPROVE ASYMPTOTIC CONVERGENCE BEHAVIOR. FOR DEPLETION, THE CODE DECIDES INTERNALLY (FOR EACH TIME-STEP AND EVERY NUCLIDE) WHETHER TO USE EITHER A DIFFERENCE APPROXIMATION OR AN ANALYTIC SOLUTION. A FULL TRIANGULAR DEPLETION MATRIX IS ALLOWED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   - 10 GROUPS
   - 20,000 SPACE MESH POINTS
   - 255 REGIONS
   - 300 SUBREGIONS
   - 40 TOTAL NUCLIDES
   - 25 HEAVY NUCLIDES
   - 25 FISSION PRODUCTS AND OTHER BURNABLE NUCLIDES

6. TYPICAL RUNNING TIME - EIGHT TIME-STEPS FOR AN X-Y PROBLEM OF 7400 MESH POINTS COULD TAKE 65 MINUTES FOR 4 GROUPS AND 155 MINUTES FOR 9 GROUPS.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - GAMTRI (ACC ABSTRACT 401) IS USED FOR THE STEADY-STATE SOLUTIONS.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   UNIVAC1108 VERSION SUBMITTED MARCH 1970.

10. REFERENCE - J. P. DORSEY, R. FROEHLICH, AND F. TODT, BUG-2/BUGTRI, TWO DIMENSIONAL MULTIGROUP BURNUP CODES FOR RECTANGULAR AND HEXAGONAL GEOMETRY, GA-8272, AUGUST 22, 1969.

11. MACHINE REQUIREMENTS - 65,536 WORDS OF CORE STORAGE, 2 MILLION WORDS OF DRUM STORAGE ON ONE DATA CHANNEL, 3 TAPE UNITS, AND A PERIPHERAL PRINTER

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASSEMBLY LANGUAGE

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC-II, GAX23.

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 (22,477 CARDS)
    SAMPLE PROBLEM (183 CARDS)
    REFERENCE REPORT

17. CATEGORY - D
    KEYWORDS - 2-DIMENSIONAL, MULTIGROUP, DIFFUSION, TRIANGULAR, HEXAGONAL, FLUX DISTRIBUTION, POWER DISTRIBUTION, DEPLETION, GAMTRI CODES

REACTOR CODE ABSTRACT 440 01/71

1. NAME OR DESIGNATION OF PROGRAM - DYNAM

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - DYNAM PERFORMS A DYNAMIC ANALYSIS OF ONCE-THROUGH BOILING FLOW OSCILLATIONS WITH STEAM SUPERHEAT. THE MODEL DESCRIBING THE SUPERHEAT REGIME (SINGLE-PHASE, VARIABLE DENSITY FLUID) FOR SUBCRITICAL PRESSURE OPERATION IS ALSO APPLICABLE TO THE STUDY OF ONCE-THROUGH OPERATION USING SUPERCRITICAL PRESSURE WATER.

4. METHOD OF SOLUTION - LINEARIZED PARTIAL DIFFERENTIAL CONSERVATION EQUATIONS ARE SOLVED USING LAPLACE TRANSFORMATION OF THE TEMPORAL TERMS AND INTEGRATION OF THE SPATIAL VARIATIONS. DYNAM IS WRITTEN IN COMPLEX VARIABLE NOTATION.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   30 INTERVALS USED TO DESCRIBE THE POWER DISTRIBUTION IN THE NONBOILING AND BOILING REGIONS
   29 BOILING NODES
   7 INTERVALS AND CORRESPONDING FRICTION MULTIPLIERS READ IN PER CASE
   14 EXIT QUALITIES READ IN PER CASE
   40 SUPERHEAT NODES
   10 COEFFICIENTS READ IN FOR THE PHI**2 VS X-POLYNOMIAL FIT
   48 FREQUENCIES AT WHICH OPEN LOOP FREQUENCY RESPONSE IS DESIRED
   48 FREQUENCIES AT WHICH SIGNAL OUTPUT IS DESIRED

6. TYPICAL RUNNING TIME - A TYPICAL PROBLEM REQUIRES 5 SECONDS.

7. UNUSUAL FEATURES OF THE PROGRAM - NO NUMERICAL INSTABILITIES ARE PRESENT.

8. RELATED AND AUXILIARY PROGRAMS - STABLE

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   UNIVAC1108 VERSION SUBMITTED MARCH 1970.

10. REFERENCES - L. E. EFFERDING, DYNAM, A DIGITAL COMPUTER PROGRAM FOR STUDY OF THE DYNAMIC STABILITY OF ONCE-THROUGH BOILING FLOW WITH STEAM SUPERHEAT, GAMD-8656, JUNE 26, 1968.
   L. E. EFFERDING, STATIC AND DYNAMIC STABILITY OF STEAM-WATER SYSTEMS - PART I, CRITICAL REVIEW OF THE LITERATURE, GA-5555, OCTOBER 1964.

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN

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

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE TEST CASE RESULTS WILL DIFFER FROM THOSE IN THE DOCUMENT GAMD-8656 BECAUSE OF THE UPDATING OF THE PRESSURE DROP EQUATIONS.

15. NAME AND ESTABLISHMENT OF AUTHORS -

ABSTRACT G. SCHLUETER

PROGRAM L. E. EFFERDING
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16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (3099 CARDS)
SAMPLE PROBLEM (33 CARDS)
REFERENCE REPORT, GAMD-8656

17. CATEGORY - H
KEYWORDS - PRESSURE, VARIATIONS, STEAM GENERATORS, HYDRODYNAMICS, TRANSFER FUNCTIONS, FLUID FLOW, STABLE CODES

REACTOR CODE ABSTRACT 441 01/71

1. NAME OR DESIGNATION OF PROGRAM - PWCOST

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - PWCOST IS USED TO CALCULATE NUCLEAR REACTOR FUEL CYCLE COSTS. INPUT FOR ALL COMPONENTS OF THE FUEL CYCLE ARE TIME-DEPENDENT. WORKING CAPITAL CHARGE RATES MAY BE SPECIFIED SEPARATELY FOR IN-CORE AND OUT-OF-CORE TIME PERIODS.

4. METHOD OF SOLUTION - COSTS ARE CALCULATED USING A PRESENT-WORTH LEVELIZED APPROACH.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   24 CORE REGIONS
   100 SEGMENTS
   2 FUEL PARTICLE TYPES
   100 RELOADS BEFORE EQUILIBRIUM IS ESTABLISHED
   100 TIMES THE CORE IS RELOADED

6. TYPICAL RUNNING TIME - 20 SECONDS ARE REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM - PWCOST CAN BE USED FOR EVALUATING COSTS OF A SINGLE FUEL SEGMENT OR FOR AN ENTIRE REACTOR HISTORY. THERE ARE SEVERAL ACCOUNTING PROCEDURES AVAILABLE AS OPTIONS INCLUDING THE TREATMENT OF PRE-IRRADIATION WORKING CAPITAL AS AN ADD-ON TO THE FRONT END SERVICES COST. THE CODE ALSO CALCULATES THE COST OF A LEASE CYCLE AND CAN EVALUATE A SPENT FUE BUYBACK OPTION.

8. RELATED AND AUXILIARY PROGRAMS - GARGOYLE (ACC ABSTRACT 260)

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   UNIVAC1108 VERSION SUBMITTED MARCH 1970.

10. REFERENCE - D. H. LEE, JR., PWCOST, A GENERAL PURPOSE COMPUTER CODE FOR THE CALCULATION OF FUEL CYCLE COSTS, GA-9394, JUNE 10, 1969.

11. MACHINE REQUIREMENTS - 65K CORE AND 1 TAPE

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 (3010 CARDS)
    SAMPLE PROBLEM (736 CARDS)
    REFERENCE REPORT

17. CATEGORY - D
    KEYWORDS - FUEL CYCLE, ECONOMICS, GARGOYLE CODES

REACTOR CODE ABSTRACT 442 01/71

1. NAME OR DESIGNATION OF PROGRAM - SIMPLE1

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - SIMPLE1 COMPILES AND EXECUTES MULTISTATEMENT CALCULATIONS TYPED IN A FAMILIAR ALGEBRAIC NOTATION AT A TIME-SHARED TERMINAL, IMMEDIATELY RETURNING REQUESTED RESULTS TO THE TERMINAL AND PERMITTING AD LIB CONTINUATION OF A CALCULATION. THE USER MAY USE THE SYSTEM AS AN EXTENDED AND POWERFUL ELECTRONIC CALCULATOR OR AS A CALCULATOR OF SMALL LOAD-AND-GO RUNS. ERRORS ARE DETECTED AND CORRECTED ON AN AS-YOU-GO BASIS. CALCULATIONS MAY INCLUDE INTERACTIVE INPUT/OUTPUT, FUNCTIONS, AND LOOPS.

4. METHOD OF SOLUTION - OPERATIONS AVAILABLE INCLUDE ALL STANDARD ARITHMETIC OPERATIONS, EXPONENTIATION, SQUARE ROOT, AND SPECIAL FUNCTIONS.

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 DECEMBER 1970.
CDC6600 VERSION SUBMITTED MARCH 1970.

10. REFERENCE - MYLES F. BARRETT, SIMPLE I, AN ALGEBRAIC PROGRAMMING LANGUAGE FOR INTERACTIVE TIME-SHARED USE, KAPL-M-7109, JANUARY 9, 1970.

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASCENT

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

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE PRESENT IMPLEMENTATION OF SIMPLE1 IS FOR THE CDC *INTERCOM* TIME-SHARING SYSTEM. THE TYPEIT SUBROUTINE IS USED IN *INTERCOM* TO LINK THE TELETYPE FOR INTERACTIVE I/O. THE PROGRAM IS SELF-CONTAINED WITH THE POSSIBLE EXCEPTION OF THE FREE FORMAT DATA READER. ANY AVAILABLE FREE FORMAT READER CAPABLE OF CONVERTING NUMERICAL INPUT TO FLOATING POINT FORM COULD BE SUBSTITUTED.

15. NAME AND ESTABLISHMENT OF AUTHOR -
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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
SOURCE DECK (681 CARDS)
REFERENCE REPORT

17. CATEGORY - P
KEYWORDS - PROGRAMMING LANGUAGES, NUMERICAL CALCULATIONS

REACTOR CODE ABSTRACT 443 01/71

1. NAME OR DESIGNATION OF PROGRAM - HAA3

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - HAA3 EVALUATES AN APPROXIMATION TO THE GENERAL INTEGRO-DIFFERENTIAL EQUATION WHICH DESCRIBES AEROSOL BEHAVIOR. THE PHYSICAL MODEL INCLUDES PARTICLE GENERATION, BROWNIAN AND GRAVITATIONAL AGGLOMERATION, SETTLING, PLATING, AND LEAKAGE FOR SPHERICAL PARTICLES. THE APPROXIMATION IS THE REQUIREMENT THAT THE SIZE DISTRIBUTION FUNCTION BE LOG-NORMAL. THE TIME-DEPENDENT PHYSICAL QUANTITIES COMPUTED ARE (1) PLATED, SETTLED, SUSPENDED, AND LEAKED VOLUME FRACTIONS, (2) GEOMETRIC MEAN DEVIATION, NUMBER CONCENTRATION, AND GEOMETRIC MEAN RADIUS FOR THE LOG-NORMAL DISTRIBUTION OF THE SUSPENDED MATERIAL, (3) THE 50 PERCENT RADIUS (BY MASS) FOR THE SUSPENDED MATERIAL, (4) THE NUMBER AND VOLUME DISTRIBUTIONS (FUNCTIONS OF RADIUS) FOR THE SETTLED, PLATED, AND LEAKED MATERIAL, (5) THE EFFECTIVE RADII FOR THE SUSPENDED, SETTLED, AND PLATED MATERIAL.

4. METHOD OF SOLUTION - THE TIME-DEPENDENT PARAMETERS WHICH CHARACTERIZE THE LOG-NORMAL SIZE DISTRIBUTION FUNCTION ARE OBTAINED BY NUMERICAL INTEGRATION OF THREE SIMULTANEOUS DIFFERENTIAL EQUATIONS FOR THE ZEROTH THROUGH THE SECOND VOLUME MOMENTS. THE VOLUME MOMENTS DIFFERENTIAL EQUATIONS WERE OBTAINED BY ANALYTIC INTEGRATION OF THE APPROPRIATE MOMENTS OF THE INTEGRO-DIFFERENTIAL EQUATION. ADDITIONAL DIFFERENTIAL EQUATIONS FOR THE PLATED, SETTLED, AND LEAKED FUNCTIONS ARE INCLUDED IN THE SIMULTANEOUS SET.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   1000 TIME-STEPS
   42 DISCRETE RADII FOR DISTRIBUTION CALCULATIONS

6. TYPICAL RUNNING TIME - 1 TO 2 MINUTES PER CASE ARE REQUIRED ON THE IBM360/50.

7. UNUSUAL FEATURES OF THE PROGRAM - THE PROGRAM WILL ALLOW ANY COMBINATION OF BROWNIAN AGGLOMERATION, GRAVITATIONAL AGGLOMERATION, SETTLING, WALL PLATING, LEAKAGE, AND LOG-NORMAL SOURCE (CONSTANT SOURCE RATE). EXTENSIVE PRINTED AND GRAPHICAL DISPLAYS ARE PROVIDED.

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   IBM360 VERSION SUBMITTED FEBRUARY 1970.

10. REFERENCE - R. S. HUBNER, AN APPROXIMATE SOLUTION TO THE GENERAL EQUATION FOR THE COAGULATION OF HETEROGENEOUS AEROSOLS, AI-AEC-MEMO-12880, SEPTEMBER 26, 1969.

11. MACHINE REQUIREMENTS - 256K BYTE IBM360 AND AN SC-4020 GRAPHICAL DISPLAY DEVICE

12. PROGRAMMING LANGUAGE USED - FORTRAN IV H

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 AIORT3 DISPLAY ROUTINE CAN BE REWRITTEN, OR A DUMMY AIORT3 CAN BE USED. THE NAR SC-4020 (OS) SUBROUTINE PACKAGE IS AVAILABLE THROUGH -

    UAIDE LIBRARIAN
    C/O STROMBERG-CARLSON
    P. O. BOX 2449
    SAN DIEGO, CALIFORNIA 92112

15. NAME AND ESTABLISHMENT OF AUTHOR -

    R. S. HUBNER
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    NORTH AMERICAN ROCKWELL
    P. O. BOX 309
    CANOGA PARK, CALIFORNIA 91304

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
    SOURCE DECK (1553 CARDS)
    SAMPLE PROBLEM (14 CARDS)
    REFERENCE REPORT

17. CATEGORY - G
    KEYWORDS - PARTICLES, DISTRIBUTION, LEAKAGE, AEROSOLS

REACTOR CODE ABSTRACT 444 01/71

1. NAME OR DESIGNATION OF PROGRAM - ROPE

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - ROPE IS USED TO CALCULATE ROOTS OF POLYNOMIALS.

4. METHOD OF SOLUTION - THREE SEPARATE METHODS ARE EMPLOYED IN FINDING THE ROOTS, THE CHOICE OF METHODS BEING A USER OPTION. THESE METHODS ARE BAIRSTOWS METHOD, LEHMERS METHOD, AND DALEMBERTS METHOD.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - ROPE CAN FIND ROOTS OF POLYNOMIALS OF ORDER UP TO 99.

6. TYPICAL RUNNING TIME - A ROUGH ESTIMATE OF RUNNING TIME IS .001N HOURS, WHERE N IS THE ORDER OF THE POLYNOMIAL.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6600 VERSION SUBMITTED MARCH 1970.

10. REFERENCES - J. R. SCHMID, R. W. GUILLETTE, AND G. O. MUELLER, ROPE, ROOTS OF A POLYNOMIAL EVALUATION, KAPL-M-6540, APRIL 1, 1966.
D. H. LEHMER, A MACHINE METHOD FOR SOLVING POLYNOMIAL EQUATIONS, JACM, VOL. 8, NO. 2, APRIL 1961.
R. F. JORDAN AND H. W. MOORE, POLRTS - BAIRSTOWS METHOD FOR FINDING ROOTS IN FORTRAN, G. E. REPORT 64TIP4, FEBRUARY 1964.
ZONE C. MOTTELER, A SIMPLE METHOD FOR FINDING THE ZEROS OF POLYNOMIALS, BASED ON DALEMBERTS LEMMA, LA-3355.

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 -

NANCY R. MITCHELL
KNOLLS ATOMIC POWER LABORATORY
GENERAL ELECTRIC COMPANY
BOX 1072
SCHENECTADY, NEW YORK 12301

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
    SOURCE DECK (705 CARDS)
    SAMPLE PROBLEM (5 CARDS)
    REFERENCE REPORT, KAPL-M-6540

17. CATEGORY - P
    KEYWORDS - POLYNOMIALS, EQUATIONS, SOLUTIONS

REACTOR CODE ABSTRACT 445 01/71

1. NAME OR DESIGNATION OF PROGRAM - LIZARD4

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - LIZARD4 IS USED TO SOLVE NON-LINEAR, ORDINARY DIFFERENTIAL EQUATIONS AS A ONE-SHOT EFFORT. IT WAS WRITTEN TO SOLVE INITIAL VALUE EQUATIONS, THAT IS, THE VALUES OF THE DEPENDENT VARIABLES MUST BE SPECIFIED AT SOME INITIAL VALUE OF THE INDEPENDENT VARIABLE. ONE-DIMENSIONAL STEADY-STATE BOUNDARY VALUE PROBLEMS CAN BE SOLVED WITH LIZARD4, PROVIDING AN ITERATIVE METHOD IS EMPLOYED BY THE USER WHERE ONE OF THE BOUNDARY CONDITIONS IS SPECIFIED AND INTEGRATION PROCEEDS UNTIL THE ALTERNATE BOUNDARY CONDITION IS MET.

4. METHOD OF SOLUTION - THE INTEGRATION PROCEDURE USED IS A MODIFIED ADAMS MOULTON METHOD COMMONLY REFERRED TO AS THE AMSINT SUBROUTINE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMUM OF - 25 NONLINEAR FIRST-ORDER DIFFERENTIAL EQUATIONS

6. TYPICAL RUNNING TIME - RUNNING TIME IS ABOUT 150 MILLISECONDS PER EQUATION PER TIME INTERVAL.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - LIZARD

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6600 VERSION SUBMITTED MARCH 1970.

10. REFERENCES - ROBERT W. GUILLETTE, GERHARD O. MUELLER, AND JOSEPH R. SCHMID, LIZARD-IV, A PROGRAM TO SOLVE NON-LINEAR, ORDINARY DIFFERENTIAL EQUATIONS, KAPL-M-6548, MARCH 17, 1966.
DAVID L. BRIGGS AND JOSEPH R. SCHMID, LIZARD, A PROGRAM TO SOLVE NON-LINEAR, ORDINARY DIFFERENTIAL EQUATIONS, KAPL-M-EC-36, OCTOBER 22, 1963.
R. J. CULLEN, 6600 CALCOMP PLOTTER ROUTINES, KAPL NOTE, JULY 25, 1966.

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASCENT

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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KNOLLS ATOMIC POWER LABORATORY

15. NAME AND ESTABLISHMENT OF AUTHOR(S) (CONTINUED)
GENERAL ELECTRIC COMPANY
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SCHENECTADY, NEW YORK 12301

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (2614 CARDS)
SAMPLE PROBLEM (39 CARDS)
REFERENCE REPORT, KAPL-M-6548, AND NOTE

17. CATEGORY - P
KEYWORDS - NONLINEAR, DIFFERENTIAL EQUATIONS, SOLUTIONS, LIZARD CODES

REACTOR CODE ABSTRACT 446 01/71

1. NAME OR DESIGNATION OF PROGRAM - MOST

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - MOST IS DESIGNED TO VARY A SET OF COORDINATES (X1,X2, . . . ,XN) REPRESENTING THE VECTOR X IN SUCH A WAY THAT A SPECIFIED FUNCTION Y(X), Y(X) GREATER THAN OR EQUAL TO 0, IS MINIMIZED.

4. METHOD OF SOLUTION - A GRADIENT SEARCH ALGORITHM IS USED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE MAXIMUM DIMENSION OF THE SEARCH IS 20.

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6600 VERSION SUBMITTED MARCH 1970.

10. REFERENCE - C. R. LUBITZ AND G. A. SHANHOLT, MOST - A MULTIDIMENSIONAL OPTIMIZATION SCHEME, KAPL-M-6174, JULY 1964.

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 -
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16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
SOURCE DECK (437 CARDS)
SAMPLE PROBLEM (6 CARDS)
REFERENCE REPORT

17. CATEGORY - P
KEYWORDS - SEARCHES, MINIMIZATION, MULTIDIMENSIONAL

REACTOR CODE ABSTRACT 447 01/71

1. NAME OR DESIGNATION OF PROGRAM - ETOG1 DATA FOR MUFT AND GAM LIBRARIES

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE DATA REQUIRED FOR THE CREATION OF MUFT4, MUFT5, GAM1 AND GAM2 LIBRARIES WERE GENERATED FROM THE BROOKHAVEN NATIONAL NEUTRON CROSS SECTION CENTER ENDF/B TAPES 114 THROUGH 117 BY THE ETOG1 PROGRAM (ACC ABSTRACT 437).

4. METHOD OF SOLUTION - THE ORIGINAL ENDF/B DATA TAPES 114, 115, AND 116 RECEIVED FROM THE BROOKHAVEN CENTER WERE ALTERED. THE CROCT PROGRAM (ACC ABSTRACT 384) WAS USED TO MAKE CORRECTIONS TO MU BAR, THE AVERAGE COSINE OF THE SCATTERING ANGLE, XI, THE LOGARITHMIC ENERGY DECREMENT, AND GAMMA, THE ANISOTROPIC GREULING-GOERTZEL PARAMETER. POINTS WERE INSERTED INTO ELASTIC SCATTERING ANGULAR DISTRIBUTION TABLES SO THAT ANISOTROPY DID NOT EXTEND DOWN TO THERMAL ENERGIES, AND THE SECONDARY DISTRIBUTION DATA FOR MATERIAL NUMBERS 1018 AND 1021 WERE CHANGED. THE MATERIALS FOR WHICH LIBRARY DATA WERE GENERATED ARE -

1001	H	1019	MN55	1037	AU197	1056	AM241
1003	D	1020	FE	1042	U233FP	1057	AM243
1005	LI6	1021	NI	1043	U234	1058	CM244
1006	LI7	1024	NB	1044	U235	1059	NA23
1007	BE9	1025	MO	1045	U235FP	1060	W182
1009	B10	1026	XE135	1046	U236	1061	W183
1010	C12	1027	SM149	1047	U238	1062	W184
1012	N14	1028	EU151	1048	NP237	1063	W186
1013	O16	1029	EU153	1050	PU238	1066	U233SFP
1014	MG	1030	GD	1051	PU239	1067	U233NFP
1015	AL27	1031	DY164	1052	PU239FP	1068	U235SFP
1016	TI	1032	LU175	1053	PU240	1069	U235NFP
1017	V51	1033	LU176	1054	PU241	1070	PU239SFP
1018	CR	1035	TA181	1055	PU242	1071	PU239NFP

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -

LIBRARY	NO. OF GROUPS
GAM1	68
GAM2	99
MUFT4,5	54

A 1/E WEIGHTING FUNCTION JOINED TO THE FISSION SPECTRUM WAS USED. A 1.0*10**7 VALUE WAS USED AS THE NON-RESONANCE POTENTIAL SCATTERING CROSS SECTION PER ABSORBER ATOM. FOR MUFT LIBRARIES THE 26TH GROUP WAS THE LOWEST GROUP NUMBER IN THE RESONANCE REGION, THE 25TH GROUP WAS THE HIGHEST IN THE INELASTIC REGION.

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - THE CRECT CODE WAS USED TO MAKE THE NECESSARY ALTERATIONS TO THE ENDF/B TAPES. THE DATA DECKS WERE CREATED FOR THE MUFT4, MUFT5, GAM1 (ACC ABSTRACT 33), AND GAM2 PROGRAMS.

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6600 VERSION SUBMITTED FEBRUARY 1970.

10. REFERENCE - S. KELLMAN AND D. E. KUSNER, DESCRIPTION OF THE GENERATION OF DATA DECKS BY ETOG-1 FOR USE IN CREATING MUFT AND GAM LIBRARIES, WCAP-3845-2 (ENDF-133), JANUARY 1970.

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED -

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 -
S. KELLMAN AND D. E. KUSNER
WESTINGHOUSE ELECTRIC CORPORATION
PWR SYSTEMS DIVISION
BOX 355
PITTSBURGH, PENNSYLVANIA 15230

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL (5 TAPES)
MUFT4 (1 TAPE 59 FILES)
MUFT5 (1 TAPE 59 FILES)
GAM1 (1 TAPE 47 FILES)
GAM2 (2 TAPES TAPE1-25 FILES, TAPE2-23 FILES)
REFERENCE REPORT

17. CATEGORY - Z
KEYWORDS - CROSS SECTIONS, LIBRARIES, MUFT4 CODES, MUFT5 CODES, GAM1 CODES, GAM2 CODES, ETOG CODES, CRECT CODES

REACTOR CODE ABSTRACT 448 01/71

1. NAME OR DESIGNATION OF PROGRAM - FLASH4

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - FLASH4 IS USED TO DETERMINE THE TRANSIENT RESPONSE OF A WATER-COOLED REACTOR OR HYDRAULIC SYSTEM TO SEVERE VARIABLE PRESSURE OPERATION.

4. METHOD OF SOLUTION - THE NUMERICAL INTEGRATIONS OF THE COUPLED FLUID CONSERVATION EQUATIONS ARE DONE BY FULLY IMPLICIT DIFFERENCE EQUATIONS. FLUID PROPERTIES ARE GENERATED FROM STORED DATA.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   20 PRESSURE DETERMINING NODES
   40 FLOW PATHS
   THE PLANT AND CORE GEOMETRY ARE ASSUMED FIXED IN TIME, HENCE THE SOLUTION IS NOT APPLICABLE AFTER SIGNIFICANT DIMENSIONAL CHANGE.

6. TYPICAL RUNNING TIME - RUNNING TIME VARIES BETWEEN .001 TO 100 TIMES REAL TIME, DEPENDING ON THE TYPE OF THE PROBLEM.

7. UNUSUAL FEATURES OF THE PROGRAM - FLASH4 CAN BE APPLIED TO ANY WATER-COOLED REACTOR GEOMETRY OR PLANT CONFIGURATION. IT INCLUDES ACOUSTIC EFFECTS FOR BOTH SUBCOOLED AND SATURATED BLOWDOWNS. THE NUMERICAL TECHNIQUES OF FLASH4 RESULT IN SHORTER RUNNING TIME FOR THE LONG-TERM SOLUTIONS OF SLOWLY VARYING TRANSIENTS, E.G. SMALL LEAKS.

8. RELATED AND AUXILIARY PROGRAMS - FLASH4 IS AN EXTENSION OF FLASH AND FLASH2 (ACC ABSTRACT 295). FLASH4 USES THE INPUT PROCESSING SUBROUTINES FCHIP, CARDS, AND INPF (SEE REFERENCE 4).

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
   CDC6600 VERSION SUBMITTED JANUARY 1970.

10. REFERENCES - T. A. PORSCHING, J. H. MURPHY, J. A. REDFIELD, AND V. C. DAVIS, FLASH-4, A FULLY-IMPLICIT FORTRAN IV PROGRAM FOR THE DIGITAL SIMULATION OF TRANSIENTS IN A REACTOR PLANT, WAPD-TM-840, MARCH 1969.
    J. A. REDFIELD, J. H. MURPHY, 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 -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

T. A. PORSCHING, J. H. MURPHY, J. A. REDFIELD,
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P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (11,598 CARDS)
SAMPLE PROBLEM (24 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS, WAPD-TM-840 AND WAPD-TM-668

17. CATEGORY - G
KEYWORDS - REACTORS, TRANSIENTS, ACCIDENT, CONTAINMENT, FUELS, FLUID FLOW, PRESSURE DISTRIBUTION, COOLANTS, BLOWDOWN, REACTOR SAFETY, FLASH CODES

REACTOR CODE ABSTRACT 449 01/71

1. NAME OR DESIGNATION OF PROGRAM - CYGRO3

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - CYGRO3 IS AN EXTENSIVE MODIFICATION OF CYGRO1 AND CYGRO2 (ACC ABSTRACT 266). BASICALLY THE PROGRAM CALCULATES TEMPERATURES, DEFORMATION, AND STRESSES IN CLADDED FUEL RODS AS A FUNCTION OF A HISTORY OF POWER AND COOLANT CONDITIONS. AXIAL AND CIRCUMFERENTIAL UNIFORMITY ARE ASSUMED. BUBBLE GROWTH AND MIGRATION ARE INCLUDED. THE MAIN CHANGES FROM CYGRO1 AND CYGRO2 ARE IN THE AREA OF VOID MIGRATION, FUEL CRACKING, CLAD COLLAPSE, REPRESENTATION OF IN-PILE CREEP AND CLAD ANISOTROPY.

4. METHOD OF SOLUTION - THE FUEL AND CLAD ARE DIVIDED INTO A NUMBER OF CONCENTRIC RINGS. EQUATIONS FOR BALANCE OF FORCES AND DISPLACEMENTS BETWEEN THE RINGS ARE USED TO ESTABLISH EQUILIBRIUM. PLASTIC FLOW (CREEP) IS TREATED ON AN INCREMENTAL BASIS. THE REQUIRED SIZES OF TIME-STEPS ARE CALCULATED INTERNALLY ON THE BASIS OF ALLOWED STRESS AND STRAIN INCREMENTS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMUM OF - 30 RINGS (FUEL PLUS CLAD)

6. TYPICAL RUNNING TIME - EACH BASIC TIME-STEP TAKES ABOUT 0.01 SECOND PER FUEL RING (I.E. 0.1 SECOND FOR 10 RINGS). THE NUMBER OF TIME-STEPS PER HISTORY INPUT STEP VARIES BETWEEN 50 AND 500.

7. UNUSUAL FEATURES OF THE PROGRAM - BENDING MOMENTS TO HOLD THE FUEL ROD STRAIGHT WHEN SITUATED IN A POWER GRADIENT ARE CALCULATED BY A PERTURBATION ANALYSIS. CORRESPONDING DEFLECTIONS IF ALLOWED TO GO FREE ARE ALSO DETERMINED. AXIAL FORCES PRODUCED BY SUPPORTS CAN BE FACTORED INTO THE CALCULATIONS. THE EFFECT OF A CENTRAL HEAT PRODUCING PLUG INSIDE THE FUEL CAN BE CALCULATED. CLAD COLLAPSE PRESSURES CAN BE SPECIFIED AS A FUNCTION OF TIME.

8. RELATED AND AUXILIARY PROGRAMS - CYGRO1 AND CYGRO2

9. STATUS - ABSTRACT FIRST DISTRIBUTED DECEMBER 1970.
CDC6600 VERSION SUBMITTED JULY 1970.

10. REFERENCES - E. DUNCOMBE AND C. M. FRIEDRICH, CYGRO-3 - A COMPUTER PROGRAM TO DETERMINE TEMPERATURES, STRESSES AND DEFORMATIONS IN OXIDE FUEL RODS (LWBR DEVELOPMENT PROGRAM), WAPD-TM-961, MARCH 1970.
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS ENVIRONMENTAL REPORT, WAPD-TM-668, JANUARY 1967.

11. MACHINE REQUIREMENTS - 140,000 OCTAL WORDS

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 - CYGRC3 CALLS A SUBROUTINE TICK WHICH IS NOT INCLUDED WITH THE PROGRAM NOR WITH THE BETTIS ENVIRONMENTAL ROUTINES. THEREFORE, THE USER WILL HAVE TO PROVIDE HIS OWN SUBROUTINE TICK. CALL TICK(HRS) IS USED TO OBTAIN ELAPSED TIME FOR THE CURRENT JOB. IT SETS HRS (REAL) TO THE ELAPSED CHARGE TIME IN HOURS.

15. NAME AND ESTABLISHMENT OF AUTHORS -

E. DUNCOMBE AND C. M. FRIEDRICH
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WESTINGHOUSE ELECTRIC CORPORATION
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL (2 TAPES)
SOURCE DECK (4760 CARDS)
SAMPLE PROBLEM (96 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS

17. CATEGORY - I
KEYWORDS - STRESSES DISTRIBUTION, FUEL ELEMENTS, DEFORMATION, TEMPERATURE, PRESSURE

VI. LIBRARY PROGRAMS BY PROBLEM CLASSIFICATION

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.

SIGMA(E(0) TO E)/SIGMA(0)=(((M+1)/M)**2) * SQRT(E/E(0)) * 1/2
THE INTEGRAL FROM -1 TO 1 OF SIGMA(E(0) TO E,COS(THETA))
D(COS(THETA))
WHERE E(0) AND E ARE THE INITIAL AND FINAL ENERGIES, THETA IS THE ANGLE OF SCATTERING, 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.

176 RAPTURE THIS CODE COMPUTES RESONANCE INTEGRALS, AVERAGES THEM OVER RESONANCE PARAMETRIC DISTRIBUTIONS, AND COMPUTES FISSION AND CAPTURE CROSS SECTIONS AS A FUNCTION OF FUEL TEMPERATURE AND OF POTENTIAL SCATTERING CROSS SECTION PER ABSORBER ISOTOPE.

177 DOPIE DOPIE WAS DEVELOPED TO STUDY THE EFFECTS OF -

(1) OVERLAP OF RESONANCES OF UNLIKE FUEL ISOTOPES SUCH AS U238 AND THE PLUTONIUM ISOTOPES, AND
(2) THE TEMPERATURE DEPENCENCE OF THE AVERAGE FLUX OVER AN ENERGY INTERVAL CONTAINING RESONANCES IN THE CALCULATION OF THE DOPPLER COEFFICIENT.

DOPIE COMPUTES FLUX-AVERAGED CROSS SECTIONS OVER A GIVEN ENERGY RANGE UNDER THE FOLLOWING ASSUMPTION

(1) INTERFERENCE SCATTERING CORRECTIONS WILL BE SMALL AND CAN BE IGNORED.
(2) SCATTERED NEUTRONS ARE DISTRIBUTED UNIFORMLY OVER THE RANGE OF ENERGY DEGRADATION CONSTANT FOR EACH MATERIAL.
(3) ALL RESONANCES ARE RESOLVED OR CAN BE TREATED AS RESOLVED.
(4) LEAKAGE IS INSIGNIFICANT, MAKING SIGMA R(J) = POTENTIAL SCATTERING.

203 COMBCO (40.0279) THE PROGRAM PERMITS COMPUTING AND CONVOLUTING A COMBINATION CROSS SECTION CURVE COMPOSED OF THE CROSS SECTION CURVES OF A SAMPLE AND UP TO FOUR CONTAMINANTS.

208 TRIX1 TRIX1 CALCULATES MULTI-GROUP, DOPPLER-BROADENED EFFECTIVE RESONANCE INTEGRALS AND CROSS SECTIONS AND THEIR TEMPERATURE COEFFICIENTS FOR A HETEROGENEOUS RESONANCE ABSORBER. USING BASIC RESONANCE PARAMETERS, GEOMETRY AND CONCENTRATIONS, THE CODE COMBINES RESOLVED RESONANCE, UNRESOLVED RESONANCE, AND 1/V AND NEGATIVE ENERGY RESONANCE CALCULATIONS. FISSILE ISOTOPE CALCULATION IS AVAILABLE. L=1 UNRESOLVED RESONANCE CALCULATIONS ARE AVAILABLE FOR ISOTOPES OF ALL SPINS. EQUIVALENCE RELATIONSHIPS ARE CODED FOR CYLINDERS, SPHERES, AND PLANES. INTERFERENCE BETWEEN RESONANCE AND POTENTIAL SCATTERING AND ITS TEMPERATURE DEPENDENCE ARE AVAILABLE. THE NARROW-RESONANCE, INFINITE-MASS, AND INTERMEDIATE REPRESENTATIONS OF SCATTERINGS WITH ABSORBER ATOMS ARE ALL AVAILABLE. THE ISOLATED SINGLE-LEVEL BREIT-WIGNER LINE SHAPE IS USED.

214 MISH MASH THE PROGRAM COMPUTES RESONANCE INTEGRALS FOR INFINITE HOMOGENEOUS MIXTURES FOR ABOVE THERMAL ENERGIES. THE MAJOR PHYSICAL APPROXIMATION IS THAT RESONANCES ARE REPRESENTED BY THE SINGLE-LEVEL BREIT-WIGNER FORMULATION. IT IS ALSO ASSUMED THAT SCATTERING IS ELASTIC AND THAT A NONRESONANT MATERIAL HAS A CONSTANT SCATTERING CROSS SECTION.

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/GEORGE CALCULATES LEGENDRE EXPANSION COEFFICIENTS FOR THE ANGULAR DISTRIBUTION OF ELASTICALLY SCATTERED NEUTRONS.

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/SQRT(E) FOR LEVELS NOT EXPLICITLY CONSIDERED. ALSO THE CROSS SECTION IN THE WINGS OF A NEGATIVE ENERGY RESONANCE, CONSTANT/(SQRT(E)*(E-EO)**2) MAY BE ADDED.

247 FLANGE1 FLANGE1 COMPUTES NEUTRON SCATTERING KERNELS FOR A LARGE CLASS OF MODERATORS. NEUTRON SCATTERING KERNELS ARE OBTAINED FROM THE SCATTERING LAW COMPUTED BY THE CODE GASKET. FLANGE1 ALLOWS THE CALCULATION OF DOUBLE DIFFERENTIAL CROSS SECTIONS, ANGULAR CROSS SECTIONS, TOTAL CROSS SECTIONS, AND LEGENDRE MOMENTS OF THE SCATTERING KERNEL. THE SHORT COLLISION TIME APPROXIMATION IS USED FOR ENERGY TRANSFERS LARGER THAN THE MAXIMUM PROVIDED BY GASKET.

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 ANISTROPIC VIBRATIONS WITH CONTINUOUS FREQUENCY SPECTRUM (AS APPLIED FOR INSTANCE TO GRAPHITE).
(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 RESONANCE PARAMETERS ARE CONSTRUCTED 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(JP1) THIS IS A PROGRAM FOR PERFORMING COUPLED-CHANNEL CALCULATIONS TO EVALUATE THE CROSS SECTIONS FOR THE SCATTERING OF NUCLEAR PARTICLES BY VARIOUS COLLECTIVE NUCLEII.

323 MUFFLE THIS PROGRAM COMPUTES THE NEUTRON CROSS SECTIONS FOR A FISSILE 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).

341 GANDY THE GANDY CODE EVALUATES TEMPERATURE-DEPENDENT EFFECTIVE NEUTRON CAPTURE, FISSION, AND SCATTERING CROSS SECTIONS IN THE UNRESOLVED RESONANCE REGION FROM AVERAGE RESONANCE PARAMETERS.

347 CODILLI A LEAST SQUARES ANALYSIS OF NEUTRON RESONANCE DATA IS PERFORMED USING THE MULTI-LEVEL EXPANSION. THE PROGRAM CAN HANDLE ONLY ONE SET OF CROSS SECTIONS AT A TIME. OPTIONS ARE PROVIDED FOR THE ANALYSIS OF REACTION OR TOTAL CROSS SECTION DATA, AND FOR THE DIRECT HANDLING OF TRANSMISSION DATA. BY OPTION, ONE CAN INCLUDE THE MULTI-LEVEL INTERFERENCE OR PERFORM THE FIT IN TERMS OF SUPERIMPOSED SYMMETRIC BREIT-WIGNER LINES, WHILE THE POTENTIAL SCATTERING INTERFERENCE IS ALWAYS INCLUDED IN THE TRIAL FUNCTION FOR THE TOTAL CROSS SECTION. PROVISIONS ARE GIVEN FOR GAUSSIAN AND NON-GAUSSIAN RESOLUTION FUNCTIONS. IN THE LATTER CASE, MODIFICATION OF ONE SUBROUTINE ALLOWS FOR ADAPTATION TO ANY KIND OF EXPERIMENTAL CONDITIONS. THE RELEVANT CONVOLUTION INTEGRALS INVOLVING THE TRIAL FUNCTION ARE EVALUATED BY SIMPSON INTEGRATION WITH AN OPTIONAL NUMBER OF INTEGRATION STEPS. BESIDES THE RESONANCES TO BE FITTED, THE TRIAL FUNCTION CONTAINS AN OPTIONAL NUMBER OF RESONANCES HAVING KNOWN PARAMETERS, WHICH MAY REPRESENT RESONANCES EXTERNAL TO THE REGION BEING FITTED AS WELL AS RESONANCES WITHIN THE ENERGY INTERVAL OF INTEREST, THUS PERMITTING IMPURITY EFFECTS TO BE DESCRIBED, OR, IF NEEDED, SPIN STATES SEPARATED.

359 PUN1 PUN1 EVALUATES UNRESOLVED RADIATIVE CAPTURE INTEGRALS AND RELATED MULTIGROUP CROSS SECTIONS. THE UNRESOLVED DISTRIBUTIONS MAY HAVE VARIOUS ORBITAL ANGULAR MOMENTUM QUANTUM NUMBERS AND THE EFFECTS OF DOPPLER BROADENING AND SELF-SHIELDING ARE INCLUDED.

360 TOR THE TOR PROGRAM CALCULATES THE SCATTERING LAW FOR A CRYSTALLINE MATERIAL IN THE INCOHERENT APPROXIMATION, FROM THE PHONON FREQUENCY DISTRIBUTION AND A SPECIFIED TEMPERATURE. A PROPORTIONAL QUANTITY IS PUNCHED IN THE FORM OF A TABLE FOR INTERPOLATION OF THE DOUBLE DIFFERENTIAL CROSS SECTION. AS AN ALTERNATIVE, THE PROGRAM OBTAINS THE ANALOGOUS QUANTITIES FOR A MONATOMIC GAS.

368 FLANGE2 FLANGE2 TAKES CROSS SECTIONS, ANGULAR DISTRIBUTION, RESONANCE PARAMETER, AND SCATTERING LAW DATA FROM ENDF/B DATA TAPES AND PREPARES THERMAL MULTIGROUP CROSS SECTIONS AND SCATTERING MATRICES.

376 AVERAGE AVERAGE CALCULATES AVERAGE SCATTERING, CAPTURE, AND FISSION CROSS SECTIONS FROM S- AND P-WAVE DATA OF THE UNRESOLVED PARAMETERS OF FILE 2 OF ENDF/B.

377 SIGPLOT SIGPLOT CALCULATES THE SCATTERING, CAPTURE, FISSION, AND TOTAL CROSS SECTIONS FROM RESOLVED RESONANCE PARAMETER DATA. SCATTERING CROSS SECTIONS MAY BE CALCULATED WITH OR WITHOUT LEVEL-LEVEL INTERFERENCE. PROVISION IS ALSO MADE TO NUMERICALLY DOPPLER-BROADEN ANY OF THE CROSS SECTIONS.

381 LYNNE LYNNE PERFORMS A MULTIPOLE EXPANSION OF THE WOODS-SAXON POTENTIAL. THE NUMBERS GENERATED ARE SUITABLE FOR MICROSCOPIC CALCULATIONS OF INELASTIC SCATTERING FROM NUCLEI WHICH USE A WOODS-SAXON INTERACTION BETWEEN THE PROJECTILE AND THE TARGET NUCLEONS.

385 COHBE/PREP THE PROGRAMS, PREP AND COHBE CALCULATE THE COHERENT ONE-PHONON SCATTERING LAW FOR POLYCRYSTALLINE BERYLLIUM, USING AN ISOTROPIC DEBYE-WALLER FACTOR AS AN APPROXIMATION FOR BERYLLIUM. THE DEBYE-WALLER FACTOR W IS CONSISTENT WITH THE DEFINITION EXP**(-2*W*ALPHA) IN THE SCATTERING LAW.

399 SUMOR(M0271) SUMOR CALCULATES S-WAVE NEUTRON CROSS SECTIONS AT SELECTED ENERGIES. THE CROSS SECTIONS ARE CALCULATED IN THREE APPROXIMATIONS TO R-MATRIX THEORY - THE REICH AND MOORE APPROXIMATION, THE FESHBACH, PORTER, AND WEISSKOPF APPROXIMATION, AND THE SUMS OF SINGLE-LEVEL FORMULAE APPROXIMATION. THESE CALCULATED CROSS SECTIONS OR, ALTERNATIVELY, INPUT LISTS OF CROSS SECTIONS, ARE DOPPLER BROADENED USING EXACT CALCULATION FOR TARGET NUCLEI IN MAXWELLIAN MOTION. CALCULATED CROSS SECTIONS OR CROSS SECTIONS DIVIDED BY NEUTRON ENERGY CAN BE INTEGRATED BETWEEN ARBITRARY ENERGY LIMITS.

410 TACASI TACASI IS USED TO DETERMINE THE PARAMETERS OF A SINGLE NEUTRON RESONANCE. THE CODE ACCEPTS MEASURED VALUES OF CAPTURE AREAS, SELF-INDICATION AREAS, SELF-INDICATION RATIOS AND TRANSMISSION AREAS AND THEIR ASSOCIATED UNCERTAINTIES IN ANY COMBINATION AND DETERMINES BEST ESTIMATES OF THE NEUTRON AND RADIATION WIDTHS AND THEIR STANDARD DEVIATIONS.

417 ATHENA4 ATHENA4 COMPUTES FORM FACTORS FOR INELASTIC SCATTERING CALCULATIONS, USING SINGLE-PARTICLE WAVE FUNCTIONS THAT ARE EIGENSTATES OF MOTION IN EITHER A WOODS-SAXON POTENTIAL WELL OR A HARMONIC OSCILLATOR WELL. TWO-BODY FORCES OF GAUSS, COULOMB, YUKAWA, AND A SUM OF CUT-OFF YUKAWA RADIAL DEPENDENCES ARE AVAILABLE.

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

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 FORM 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 CYLINDRICAL GEOMETRY CELL 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 ZOT 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.

B. SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS 2/71

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 DANCOFF 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 DANCOFF 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 SOPHIST 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 VERSION) 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.

B. SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS 2/71

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 CDC1604B 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.

B. SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS 2/71

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.C7EV. 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.

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 DANCOFF 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 ARBITRAR TABLE OF VALUES.

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 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 IMMERSED IN A HOMOGENEOUS MEDIUM.

276 AVOID 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.

B. SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS 2/71

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

280 MO807 MO807 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.

285 RESQ2/RESQO/CBF1 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 PO 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.

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. THESE OPTIONS ARE DESCRIBED IN REFERENCE 2.

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.

B. SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS 2/71

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 GAFGAR/P3T/PROC/TAPCOP 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.

355 MC**2 MC**2 IS USED TO CALCULATE MULTIGROUP CROSS SECTIONS USING AN EVALUATED NUCLEAR DATA FILE (ENDF) AND THESE CROSS SECTIONS ARE SUITABLE FOR DIRECT USE BY NEUTRONICS CODES WITHOUT PERFORMING ANCILLARY CALCULATIONS.

361 GLEN THE GLEN PROGRAM INTERPOLATES VALUES OF A FACTOR PROPORTIONAL TO THE SCATTERING LAW FROM THE PUNCHED OUTPUT OF THE TOR CODE (ACC ABSTRACT 360). THE DIFFERENTIAL CROSS SECTION DETERMINED FROM THESE IS INTEGRATED OVER THE SCATTERING ANGLE TO OBTAIN COEFFICIENTS OF AN EXPANSION IN LEGENDRE POLYNOMIALS OF THIS ANGLE FOR L = 0, 1, 2, 3. INTEGRATION OVER FINAL ENERGIES YIELDS VALUES OF THE TOTAL SCATTERING CROSS SECTION AND TRANSPORT CROSS SECTION. FOR EACH OF A SERIES OF ISOTOPIC COMPOSITIONS (UP TO 10 COMPOSITIONS) THE GLEN CODE CALCULATES THE DIFFUSION LENGTH AND VALUES OF THE FLUX-WEIGHTED GROUP AVERAGE MACROSCOPIC SCATTERING, ABSORPTION, FISSION, AND TRANSFER CROSS SECTIONS.

362 WELWING WELWING WAS DEVELOPED TO CALCULATE THE MATERIAL BUCKLING OF REACTOR SYSTEMS CONSISTING OF ANNULAR FUEL ELEMENTS IN HEAVY WATER AS MODERATOR FOR VARIOUS MODERATOR TO FUEL RATIOS. THE MODERATOR TO FUEL RATIO FOR THE MAXIMUM MATERIAL BUCKLING FOR THE PARTICULAR SYSTEM IS SELECTED AUTOMATICALLY AND THE CORRESPONDING MATERIAL BUCKLING IS CALCULATED.

374 1DX 1DX IS A MULTIPURPOSE, ONE-DIMENSIONAL DIFFUSION CODE FOR GENERATING CROSS SECTIONS TO BE USED IN FAST REACTOR ANALYSES. THE CODE IS DESIGNED TO -

(A) COMPUTE AND PUNCH RESONANCE SHIELDED CROSS SECTIONS USING DATA IN THE RUSSIAN (SEE REFERENCE 2) FORMAT,
(B) COMPUTE AND PUNCH GROUP-COLLAPSED MICROSCOPIC AND/OR MACROSCOPIC CROSS SECTIONS AVERAGED OVER THE SPECTRUM IN ANY SPECIFIED ZONE, AND
(C) COMPUTE KEFF AND PERFORM CRITICALITY SEARCHES ON TIME ABSORPTION, MATERIAL CONCENTRATIONS, ZONE DIMENSIONS, AND BUCKLING USING EITHER A FLUX OR AN ADJOINT MODEL.

388 ETOX ETOX (ENDF/B TO 1DX) CALCULATES MULTIGROUP CONSTANTS FOR NUCLEAR REACTOR CALCULATIONS USING DATA FROM THE EVALUATED NUCLEAR DATA FILE (ENDF/B). THE CODE IS DESIGNED TO COMPUTE AND PUNCH -

(A) INFINITE DILUTE CROSS SECTIONS,
(B) TEMPERATURE DEPENDENT SELF-SHIELDING FACTORS FOR ARBITRARY VALUES OF MICROSCOPIC SIGMAO (TOTAL CROSS SECTION PER ATOM) IN THE RUSSIAN (BONDARENKO) FORMAT, AND
(C) INELASTIC SCATTERING PROBABILITY MATRICES.

392 RAFFLE RAFFLE CALCULATES NEUTRON FIRST FLIGHT COLLISION PROBABILITIES FOR A WIDE VARIETY OF THREE-DIMENSIONAL CELL GEOMETRY CONFIGURATIONS. THE OUTER BOUNDARIES OF THE CELL CROSS SECTION MAY BE CIRCULAR, SQUARE, OR HEXAGONAL. THE CELL MAY CONTAIN ANNULAR REGIONS AND/OR CLUSTERS OF RODS.

393 XSDRN XSDRN USES THE NORDHEIM INTEGRAL TREATMENT, NARROW RESONANCE, OR INFINITE MASS APPROXIMATION TO PROCESS RESONANCE DATA ON A MASTER CROSS SECTION LIBRARY AND THUS OBTAIN MICROSCOPIC FINE-GROUP CROSS SECTIONS FOR A LARGE NUMBER OF NUCLIDES. THE CODE WILL THEN USE THESE CROSS SECTIONS IN AN INDEPENDENT CALCULATION TO SOLVE FOR FLUXES, EIGENVALUES, CRITICAL DIMENSIONS, ETC., USING DISCRETE ORDINATES, DIFFUSION, OR AN INFINITE MEDIUM THEORY CALCULATION. THE FINE-GROUP FLUXES THUS OBTAINED CAN THEN BE USED TO COLLAPSE THE FINE-GROUP CROSS SECTION DATA TO A MORE TENABLE BROAD-GROUP STRUCTURE FOR USE IN SEVERAL INDEPENDENT COMPUTER CODES.

416 PARTI PARTI IS A GROUP COLLAPSING CODE WHICH DETERMINES THE OPTIMUM DISCRETE REPRESENTATION OF A VARIABLE FOR SUBSEQUENT REPETITIVE CALCULATIONS.

B. SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS 2/71

420 GROUSE GROUSE COMPUTES EFFECTIVE MULTIGROUP CROSS SECTIONS AS A FUNCTION OF POSITION IN THE CORE OF A REACTOR OR THE ABSORBER REGION OF A FUEL OR CONTROL ELEMENT. THE FLUX WEIGHTING USES SYNTHESIZED FLUXES GENERATED IN FINE ENERGY DETAIL FROM PARAMETRIC FLUX TRAVERSES SUPPLIED AS INPUT AS A FUNCTION OF THE CORE ABSORPTION AND SCATTER CROSS SECTIONS.

426 PAX02 PAX02 IS USED TO GENERATE A HARMONY FILE CONTAINING FEW-GROUP MICROSCOPIC AND/OR MACROSCOPIC CROSS SECTIONS FOR PDQ7 PROBLEMS. INFINITE MEDIUM, TIME-DEPENDENT CALCULATIONS OF NEUTRON FLUX SPECTRA IN HOMOGENEOUS, HYDROGENOUS MEDIA ARE PERFORMED. HETEROGENEOUS RESONANCE INTEGRALS, THERMAL SHIELDING FACTORS, AND BLACKNESS PARAMETERS ARE CALCULATED. COEFFICIENTS FOR TEMPERATURE FEEDBACK CALCULATIONS ARE OBTAINED.

431 SUPERTOG SUPERTOG ACCEPTS NUCLEAR DATA IN EITHER A POINT BY POINT OR PARAMETRIC REPRESENTATION AS SPECIFIED BY ENDF/B. THIS DATA IS AVERAGED OVER EACH SPECIFIED GROUP WIDTH. THE EXPLICIT ASSUMPTION IS MADE THAT THE FLUX PER UNIT LETHARGY IS CONSTANT OR THAT A SUITABLE WEIGHT FUNCTION WILL BE SUPPLIED BY THE USER. WHEN RESONANCE DATA IS AVAILABLE, RESOLVED AND UNRESOLVED RESONANCE CONTRIBUTIONS ARE CALCULATED AND USED AS SPECIFIED BY INPUT OPTIONS. FINE GROUP CONSTANTS SUCH AS ONE-DIMENSIONAL REACTION ARRAYS (ABSORPTION, FISSION, ETC.), PN ELASTIC SCATTERING MATRICES, AND INELASTIC AND (N,2N) SCATTERING MATRICES ARE GENERATED AND PLACED ON TAPES IN FORMATS SUITABLE FOR USE BY THE GAM1 (ACC ABSTRACT 33), GAM2, ANISN, OR DOT PROGRAMS.

436 ETOM1 ETOM1 PROCESSES BASIC NUCLEAR INFORMATION GIVEN IN THE ENDF/B FORMAT AND PRODUCES DATA DECKS FOR USE IN GENERATION OF MUFT4 AND MUFT5 LIBRARIES.

437 ETOG1 ETOG1 PROCESSES BASIC NUCLEAR INFORMATION GIVEN IN THE ENDF/B FORMAT AND PRODUCES DATA DECKS FOR USE IN GENERATION OF MUFT4, MUFT5, GAM1, GAM2, AND ANISN LIBRARIES.

7 FIRN FIRN 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. FIRN IS LIMITED TO A MAXIMUM OF SIX GROUPS AND THE S2, S4 OR S6 APPROXIMATION.

9 FIRE5 FIRE5 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. FIRE5 CAN BE USED FOR CELL CALCULATIONS WITH THE BOUNDARY CONDITION THAT THE CURRENT BE ZERO ON THE OUTER BOUNDARY. THE EFFECTIVE CROSS SECTIONS AND DIFFUSION COEFFICIENTS FOR A HOMOGENISED REGION CONSISTING OF ANY NUMBER OF NEIGHBORING REGIONS CAN ALSO BE OBTAINED, BUT ONLY FOR CYLINDRICAL GEOMETRY. FIRE5 ALSO COMPUTES THE MACROSCOPIC FLUX-WEIGHTED CROSS SECTIONS FOR COLLAPSED GROUPS IN EACH REGION.

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 TSCHEBYSCHEFF 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 FCG 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 EQUIPOISE-3A 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

(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 TECHNICIAN 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 TDP 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, MANY GROUP DSN 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 GAMBLE4/GAMBLE5 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.

225 TEMCO TEMCO COMPUTES REACTOR TEMPERATURE COEFFICIENTS.

241 HFN HFN SOLVES THE HOMOGENEOUS OR INHOMOGENEOUS ONE-DIMENSIONAL MULTIGROUP DIFFUSION EQUATION FOR IT 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, 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 KEFF OR ALPHA, INVERSE PERIOD, VALUE MAY BE SPECIFIED RATHER THAN THE CRITICALITY VALUE, KEFF = 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.

342 MO648 MO648 SOLVES THE ONE-DIMENSIONAL SLAB TRANSPORT PROBLEM WITH SLOWING DOWN FOR AN ARBITRARY SPATIAL EXTERNAL SOURCE AND ARBITRARY SCATTERING.

358 TWOTRAN TWOTRAN SOLVES TWO-DIMENSIONAL PARTICLE TRANSPORT PROBLEMS. SEPARATE VERSIONS ARE AVAILABLE IN X-Y AND R-Z GEOMETRIES AS WELL AS A GENERAL GEOMETRY VERSION IN X-Y, R-Z, AND R-THETA GEOMETRIES. BOTH DIRECT AND ADJOINT, HOMOGENEOUS (KEFF OR PARAMETRIC EIGENVALUE SEARCHES) OR INHOMOGENEOUS TIME-INDEPENDENT PROBLEMS ARE SOLVED SUBJECT TO VACUUM, REFLECTIVE, OR INPUT SPECIFICATION OF BOUNDARY FLUX CONDITIONS. BOTH ANISOTROPIC INHOMOGENEOUS PROBLEMS AND GENERAL ANISOTROPIC SCATTERING PROBLEMS ARE TREATED.

380 GATT GATT IS A THREE-DIMENSIONAL FEW-GROUP NEUTRON DIFFUSION THEORY PROGRAM FOR CALCULATING THE DETAILED SPATIAL FLUX AND POWER DISTRIBUTION FOR REACTORS WITH HEXAGONAL CORE CONFIGURATION. THE PROGRAM USES A UNIFORM TRIANGULAR MESH IN THE HORIZONTAL MESH PLANES AND ASSUMES A RELATIVELY SIMPLE REGION STRUCTURE IN THE AXIAL DIRECTION. IT WAS DESIGNED TO REPRESENT THE SPECIAL PATCH-TYPE CORE STRUCTURE OF THE HTGR REACTOR AS CLOSELY AS POSSIBLE.

398 BE21 BE21 SOLVES THE FEW-GROUP DISCRETE ORDINATES EQUATIONS IN SLAB GEOMETRY. EITHER HOMOGENEOUS OR INHOMOGENEOUS PROBLEMS CAN BE SOLVED.

401 GAMTRI THE HOMOGENEOUS TWO-DIMENSIONAL MULTIGROUP DIFFUSION THEORY EQUATIONS WITH ARBITRARY GROUP-TO-GROUP SCATTERING AND ARBITRARY FISSION TRANSFER ARE SOLVED FOR HETEROGENEOUS ASSEMBLIES IN (UNIFORM MESH) TRIANGULAR GEOMETRY. HOMOGENEOUS LOGARITHMIC BOUNDARY CONDITIONS ARE USED AT THE OUTER SURFACE OF NON-DIFFUSION REGIONS. THE RESULTS INCLUDE THE GROUP AND POINT-DEPENDENT NEUTRON FLUXES, THE POWER DISTRIBUTION, THE NEUTRON MULTIPLICATION FACTOR, AND A DETAILED NEUTRON BALANCE.

43C GAZE2 GAZE2 IS A ONE-DIMENSIONAL, MULTIGROUP, NEUTRON DIFFUSION THEORY PROGRAM. IT INCLUDES ALL FOUR OF THE STANDARD ONE-DIMENSIONAL GEOMETRIES - SLAB, SPHERE, RADIAL CYLINDER, AND AXIAL CYLINDER, THE LAST OF WHICH IS IDENTICAL TO SLAB GEOMETRY EXCEPT WHEN THE ZOOM ALBEDO-TYPE TRANSVERSE BOUNDARY CONDITION IS USED.

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

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 INTERVALS, WITH EXPOSURE TO A TWO-GROUP NEUTRON FLUX. INFINITE SYSTEM CRITICALITY IS CALCULATED.

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

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 ISOTOPES 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.

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

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.

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 MULTIGROUP, TWO-DIMENSIONAL REACTOR DEPLETION. GIVEN NUCLIDE CONCENTRATIONS AND MICROSCOPIC CROSS SECTIONS, THE STEADY-STATE MULTIREGION, MULTIGROUP DIFFUSION EQUATIONS ARE SOLVED IN ONE OR TWO DIMENSIONS OVER A FINITE-DIFFERENCE SYSTEM OF MESH POINTS. THE 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.

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

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.

275 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.

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

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 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 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.

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

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 ISOSEARCH 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.

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

(A) COMPUTE FUEL BURNUP USING A FLEXIBLE MATERIAL SHUFFLING SCHEME,

(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,

(C) COMPUTE FLUX DISTRIBUTIONS FOR AN ARBITRARY EXTRANEOUS SOURCE.

336 PDQ5 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.

339 GAUGE THE TWO-DIMENSIONAL FEW-GROUP NEUTRON DIFFUSION THEORY EQUATIONS FOR A UNIFORM TRIANGULAR MESH ARE SOLVED TO OBTAIN THE MULTIPLICATION FACTOR AND THE SPATIAL FLUX AND POWER DISTRIBUTION OF REACTORS WITH HEXAGONAL CORE CONFIGURATION. COMPLETE REACTOR LIFE HISTORIES WITH PARTIAL REFUELLING AT A NUMBER OF RELOAD TIME POINTS CAN BE CALCULATED. AT EACH DISCRETE TIME POINT A CONTROL ROD SEARCH MAY BE PERFORMED TO MAINTAIN CRITICALITY AT ALL TIMES. THE DEPLETION SCHEME OF ALL BURNABLE NUCLIDES IS SPECIFIED BY THE USER AT EXECUTION TIME. THREE MODES OF OPERATION ARE POSSIBLE - (1) STRAIGHT BURNUP CALCULATION, (2) CONTROL ROD CRITICALITY SEARCH, ALLOWING THE ADJUSTMENT OF A NUMBER OF CONTROL ROD BANKS ACCORDING TO A PRESCRIBED ROD SEQUENCING SCHEME, AND (3) A SERIES OF STATIC CALCULATIONS WITH INSERTION OF RODS INTO FIXED PRESCRIBED POSITIONS.

C. DEPLETION, FUEL MANAGEMENT, COST ANALYSIS, AND REACTOR ECONOMICS 2/71

340 POWERCO POWERCO CALCULATES THE COST OF ELECTRICITY PRODUCED BY NUCLEAR POWER STATIONS, ASSUMING ALL CASH EXPENSES SUCH AS INVESTMENT AND FUEL COSTS, OPERATING EXPENSES, AND TAXES ARE KNOWN. THE POWER COST IS HELD CONSTANT THROUGHOUT THE PROJECT LIFE.

354 CINCAS CINCAS IS A NUCLEAR FUEL CYCLE COST CODE WHICH MAY BE USED FOR EITHER ENGINEERING ECONOMY PREDICTIONS OF FUEL CYCLE COSTS OR FOR ACCOUNTING FORECASTING OF SUCH COSTS. FEATURES OF CINCAS INCLUDE -

(1) MONTHLY CALCULATION OF DOLLAR COSTS AND MASS INVENTORY ON A BATCH AND CASE BASIS FOR EACH MONTH OF A PERIOD WHICH IS USUALLY DEFINED AS (BUT NOT RESTRICTED TO) BEGINNING WITH THE DELIVERY OF FUEL TO THE REACTOR SITE AND ENDING WITH THE WITHDRAWAL OF FUEL FROM THE REACTOR.

(2) A GENERAL FORMULA FOR THE UNIT PRICE OF ENRICHED URANIUM WHICH ALLOWS FOR VARIABLE FEED AND TAILS ENRICHMENTS, COSTS OF FEED, CHEMICAL CONVERSION, SEPARATIVE WORK, AND LOSSES IN CONVERSION AND FABRICATION.

367 ISOGEN ISOGEN CALCULATES RADIOISOTOPE GENERATION AND DECAY, USING TWO-GROUP NEUTRON CROSS SECTIONS.

372 RAPFU RAPFU CALCULATES EQUILIBRIUM FUEL CYCLE ISOTOPICS IN FAST BREEDER REACTORS. THE RECYCLED PLUTONIUM IS PERMITTED TO HAVE DIFFERENT ISOTOPIC COMPOSITIONS IN TWO DIFFERENT CORE ZONES, AND SEVERAL RECYCLE SCHEMES ARE AVAILABLE AS OPTIONS. OUTPUT DATA INCLUDES THE INITIAL, AVERAGE, AND DISCHARGED FUEL ISOTOPIC CONCENTRATIONS FOR EACH REGION OF THE CORE ZONES AND THE BLANKETS, BREEDING RATIO, DOUBLING TIME, AND (OPTIONALLY) FUEL COSTS CALCULATED USING SIMPLIFIED RELATIONSHIPS.

418 CHAINS CHAINS COMPUTES THE ATOM DENSITY OF MEMBERS OF A SINGLE RADIOACTIVE DECAY CHAIN. THE LINEARITY OF THE BATEMAN EQUATIONS ALLOWS TRACING OF INTERCONNECTING CHAINS BY MANUALLY ACCUMULATING RESULTS FROM SEPARATE CALCULATIONS OF SINGLE CHAINS. RE-ENTRANT LOOPS CAN BE TREATED AS EXTENSIONS OF A SINGLE CHAIN. LOSSES FROM THE CHAIN ARE ALSO TALLIED.

423 DOS DOS COMPUTES THE LOCAL NEUTRON FLUX PER REFERENCE REACTOR POWER LEVEL WHICH WILL PRODUCE THE MEASURED ACTIVITY OF A DOSIMETER. ALTERNATIVELY, IF THE FLUX VALUE IS SUPPLIED, THE CORRESPONDING DOSIMETER ACTIVITY WILL BE CALCULATED.

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

427 FARED FARED CONTAINS AN INTERNAL CROSS SECTION AVERAGING ROUTINE WHICH IS RESPONSIBLE FOR PREPARING BROAD GROUP CROSS SECTION SETS FOR VARIOUS MATERIAL REGIONS OF THE REACTOR. THE CROSS SECTION AVERAGING IS PERFORMED IN PROGRAM REGA, WHICH COMPUTES A B1 FLUX AND CURRENT IN UP TO 20 REACTOR BLOCK COMPOSITIONS FOR USE AS WEIGHTING FUNCTIONS IN THE CROSS SECTION COLLAPSING CALCULATION. A HOMOGENEOUS OR HETEROGENEOUS RESOLVED AND UNRESOLVED RESONANCE TREATMENT IS PROVIDED TO COMPUTE EFFECTIVE MICROGROUP RESONANCE CROSS SECTIONS FOR THE BLOCK MIXTURE OR UP TO 2 CELL TYPES PER BLOCK. REAL AND ADJOINT FLUX DISTRIBUTIONS ARE CALCULATED FOR ONE-DIMENSIONAL SLAB, CYLINDRICAL, OR SPHERICAL GEOMETRIES. THE REAL FLUXES ARE NORMALIZED TO YIELD DESIRED TOTAL REACTOR POWER. CRITICALITY SEARCHES MAY BE PERFORMED ON THE REACTOR DIMENSION, TRANSVERSE BUCKLING OR ZONE COMPOSITIONS. ENRICHMENT SEARCHES MAY BE PERFORMED TO YIELD DESIRED RATIOS OF MAXIMUM (OR AVERAGE) POWER DENSITIES IN SEVERAL ZONES. ZONEWISE DEPLETION IS CALCULATED EITHER FOR A GIVEN TIME PERIOD OR UNTIL SPECIFIED CRITICALITY, BURNUP OR NUCLIDE CONCENTRATIONS ARE SATISFIED. FLEXIBLE FUEL MANAGEMENT IS AVAILABLE PERMITTING SPECIFIED MATERIAL UNITS TO BE MOVED INTO, OUT OF OR SHUFFLED WITHIN THE REACTOR. A WIDE VARIETY OF EDITS MAY BE PERFORMED, INCLUDING PERTURBATION AND KINETIC PARAMETERS CALCULATIONS.

438 BUG2 THE TWO-DIMENSIONAL MULTIGROUP NEUTRON DIFFUSION THEORY EQUATIONS FOR X-Y OR R-Z GEOMETRY ARE SOLVED TO OBTAIN THE MULTIPLICATION FACTOR AND THE SPATIAL FLUX AND POWER DISTRIBUTIONS. COMPLETE REACTOR LIFE HISTORIES WITH PARTIAL REFUELING AT A NUMBER OF RELOAD POINTS CAN BE CALCULATED. THE DEPLETION SCHEME OF ALL BURNABLE NUCLIDES IS SPECIFIED BY THE USER AT EXECUTION TIME. A REGIONWIDE DEPLETION SCHEME IS USED. CONCENTRATION DEPENDENT SELF-SHIELDING FACTORS MAY BE APPLIED TO ANY NUCLIDE.

439 BUGTRI THE TWO-DIMENSIONAL MULTIGROUP NEUTRON DIFFUSION THEORY EQUATIONS FOR TRIANGULAR GEOMETRY ARE SOLVED TO OBTAIN THE MULTIPLICATION FACTOR AND THE SPATIAL FLUX AND POWER DISTRIBUTIONS. COMPLETE REACTOR LIFE HISTORIES WITH PARTIAL REFUELING AT A NUMBER OF RELOAD TIME POINTS CAN BE CALCULATED. THE DEPLETION SCHEME OF ALL BURNABLE NUCLIDES IS SPECIFIED BY THE USER AT EXECUTION TIME. A REGIONWISE DEPLETION SCHEME IS USED. CONCENTRATION DEPENDENT SELF-SHIELDING FACTORS MAY BE APPLIED TO ANY NUCLIDE.

441 PWCOST PWCOST IS USED TO CALCULATE NUCLEAR REACTOR FUEL CYCLE COSTS. INPUT FOR ALL COMPONENTS OF THE FUEL CYCLE ARE TIME-DEPENDENT. WORKING CAPITAL CHARGE RATES MAY BE SPECIFIED SEPARATELY FOR IN-CORE AND OUT-OF-CORE TIME PERIODS.

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 PRECURSORS ARE DISPLAYED GRAPHICALLY.

122 SNAPKIN5/SNAPKIN-5A 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, WEIGHTS POWER, HEAT CAPACITY, AND REACTIVITY IMPORTANCE FOR TWENTY-FIVE OR FEWER REGIONS.

135 TRAFICORPORATION, COMPLEX TRA 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 (GENERAL ATOMIC STABILIT 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.

303 BLOOST6 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.

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 X 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.

363 BLAST BLAST HAS BEEN DEVELOPED TO STUDY ACCIDENT CONDITIONS IN CRITICAL AND SUBCRITICAL THERMAL MULTIPLYING SYSTEMS. THE PROGRAMME COMPUTES THE TIME BEHAVIOUR OF THE THERMAL NEUTRON DENSITY AND THE SYSTEM TEMPERATURE FOLLOWING A STEP CHANGE IN REACTIVITY. THE INTEGRATED THERMAL NEUTRON DENSITY IS ALSO COMPUTED, FROM WHICH THE TOTAL NUMBER OF FISSIONS DURING AN EXCURSION MAY BE OBTAINED.

F. SPACE-TIME KINETICS, COUPLED NEUTRONICS-HYDRODYNAMICS-THERMODYNAMICS, AND EXCURSION SIMULATIONS 2/71

102 AX1 AX1 PERFORMS A COUPLED NEUTRONICS-HYDRODYNAMICS CALCULATION. GIVEN A SPHERICALLY SYMMETRIC, SUPERPROMPT 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.

F. SPACE-TIME KINETICS, COUPLED NEUTRONICS-HYDRODYNAMICS-THERMODYNAMICS, AND EXCURSION SIMULATIONS 2/71

174 FORE/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.

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.

274 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.

F. SPACE-TIME KINETICS, COUPLED NEUTRONICS-HYDRODYNAMICS-THERMODYNAMICS, AND EXCURSION SIMULATIONS 2/71

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 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.

F. SPACE-TIME KINETICS, COUPLED NEUTRONICS-HYDRODYNAMICS-THERMODYNAMICS, AND EXCURSION SIMULATIONS 2/71

338 TWIGL/TWIGGLE TWIGL SOLVES THE TWO-DIMENSIONAL, TWO-GROUP, SPACE-TIME NEUTRON DIFFUSION EQUATIONS IN RECTANGULAR OR CYLINDRICAL GEOMETRY IN THE PRESENCE OF TEMPERATURE FEEDBACK. THE NEUTRON DIFFUSION AND DELAYED PRECURSOR EQUATIONS ARE DIFFERENCED IN BOTH SPACE AND TIME. THE THERMAL-HYDRAULIC DESCRIPTION IS BASED ON A NO-BOILING, ONE-PASS MODEL FORMULATED IN TERMS OF REGIONWISE-AVERAGED COOLANT AND FUEL METAL TEMPERATURES.

352 RAUMZEIT RAUMZEIT SOLVES SYSTEMS OF ONE-DIMENSIONAL, TIME-DEPENDENT, MULTIGROUP DIFFUSION-TYPE EQUATIONS USING EITHER OF TWO TREATMENTS OF THE TIME DEPENDENCE, FINITE DIFFERENCING OR THE TIME-INTEGRATED APPROACH (SEE REFERENCE 2).

370 GAKIT GAKIT SOLVES THE MULTIGROUP, ONE-DIMENSIONAL, TIME-DEPENDENT DIFFUSION THEORY KINETICS EQUATIONS INCLUDING DELAYED NEUTRON EFFECTS AND TEMPERATURE FEEDBACK BASED ON TWO-DIMENSIONAL HEAT TRANSFER CALCULATIONS. FOR THE ONE-DIMENSIONAL MULTIGROUP KINETICS EQUATIONS AN ARBITRARY SCATTERING MATRIX AND ARBITRARY FISSION TRANSFER ARE ALLOWED, AND PLANE, CYLINDRICAL, OR SPHERICAL GEOMETRY MIGHT BE USED. A PIECEWISE LINEAR TIME-DEPENDENT INHOMOGENEOUS SOURCE CAN BE SPECIFIED. FEEDBACK IS AVAILABLE FROM XENON BUILDUP AND TEMPERATURE DEPENDENCE OF CROSS SECTIONS. THE HEAT TRANSFER CALCULATION IS PERFORMED FOR TWO-DIMENSIONAL R-Z FUEL ELEMENT MODELS ASSUMING PREDETERMINED AXIAL POWER SHAPE FUNCTIONS AND TIME-DEPENDENT POWER AMPLITUDES OBTAINED FROM THE ONE-DIMENSIONAL KINETICS CALCULATIONS. FOR THE FUEL ELEMENTS AVERAGE FUEL AND MODERATOR TEMPERATURES ARE CALCULATED WHICH DETERMINE, BASED ON TABLES, THE TEMPERATURE-DEPENDENT CROSS SECTIONS. TRANSIENTS MAY BE INTRODUCED BY STEP CHANGES OF CROSS SECTIONS, BY PIECEWISE LINEAR TIME-DEPENDENT CROSS SECTIONS (ROD WITHDRAWAL ACCIDENTS), BY STEP CHANGES OF THE FLOW RATES OR BY STEP CHANGES OF THE COOLANT INLET TEMPERATURES.

371 NOWIG NOWIG IS A PROGRAM FOR SOLVING THE ONE-DIMENSIONAL TWO-GROUP NEUTRON DIFFUSION AND DELAYED PRECURSOR EQUATIONS USING A SHAPE-SPECIFIED POINT KINETICS APPROXIMATION. FEEDBACK DUE TO CHANGES IN THE FUEL METAL TEMPERATURE AND COOLANT DENSITY IS ACCOUNTED FOR BY USING A MODEL WHICH IS IDENTICAL WITH THAT USED IN THE WIGL2 (ACC ABSTRACT 274) PROGRAM.

389 STINT3 STINT3 SOLVES STATIC (EIGENVALUE) AND TIME-DEPENDENT SYSTEMS OF COUPLED, ONE-DIMENSIONAL, DIFFUSION TYPE EQUATIONS IN SLAB GEOMETRY AND IS PRIMARILY INTENDED FOR SOLVING SINGLE-CHANNEL, FLUX-SYNTHESIS EQUATIONS. THE CODE PROVIDES FOR CONTROL ROD MOTION AND TEMPERATURE FEEDBACK.

F. SPACE-TIME KINETICS, COUPLED NEUTRONICS-HYDRODYNAMICS-THERMODYNAMICS, AND EXCURSION SIMULATIONS 2/71

400 SAS1A SAS1A IS USED FOR THE ANALYSIS OF FAST REACTOR POWER AND FLOW TRANSIENTS. THE PROGRAM CONSISTS OF FOUR DRIVER PROGRAMS EACH CALLED BY THE MAIN PROGRAM. THESE DRIVERS HANDLE THE AREAS OF (1) INPUT/OUTPUT, WITH DIAGNOSTICS, (2) STEADY-STATE INITIALIZATION, TO DEFINE THE CORE OPERATING CONDITIONS BEFORE THE INITIATION OF THE TRANSIENT, (3) THE TRANSIENT PORTION, AND (4) THE DISASSEMBLY (WEAK EXPLOSION) ANALYSIS.

405 NOAH NOAH SOLVES THE ONE-DIMENSIONAL, ONE-GROUP SPACE-TIME DIFFUSION EQUATION ACCOUNTING FOR THE EFFECTS OF FUEL, CLAD, AND COOLANT TEMPERATURES (OR BY CHANGING SUBROUTINES FUEL, COOLANT, AND SOLID MODERATOR TEMPERATURES) ON FISSION AND ABSORPTION CROSS SECTIONS, AND ON THE DIFFUSION COEFFICIENT AND THE TRANSVERSE BUCKLING. IT CAN ACCOUNT FOR THE EFFECTS OF XENON-IODINE FEEDBACK. IF DESIRED, IT WILL DETERMINE LONG-TIME XENON-FLUX BEHAVIOR ASSUMING THE TEMPERATURES TO BE IN QUASI-STATIC EQUILIBRIUM. NUMEROUS METHODS OF PERTURBATION ARE ALLOWED AND CONTROL OF THE TRANSIENT IS ALSO PROVIDED.

415 CEXE/INCEXE CEXE SOLVES THE THREE-DIMENSIONAL XYZ TIME-DEPENDENT XENON SPATIAL OSCILLATION PROBLEM USING A MODIFIED ONE ENERGY GROUP THEORY AND A NODAL REPRESENTATION.

47 CLOUD 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 BLOWDOWN2 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. 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.

284 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.

295 FLASH3 FLASH3 DETERMINES THE TRANSIENT RESPONSE OF A WATER-COOLED 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.

366 CHEMLOC2 CHEMLOC2 IS USED TO COMPUTE THE EXTENT OF CORE HEATING AND METAL-STEAM REACTION FOLLOWING A HYPOTHETICAL LOSS-OF-COOLANT ACCIDENT, INCLUDING EMERGENCY CORE-COOLING FAILURE, IN A WATER-COOLED REACTOR. THE PROGRAM INCLUDES THE EFFECTS OF HEAT PRODUCTION BY DECAY HEATING IN THE FUEL AS WELL AS BY STEAM-METAL CHEMICAL REACTION, HEAT TRANSFER FROM THE CORE TO THE BOTTOM AND TOP GRID-SUPPORT PLATES AND TO THE WALL SURROUNDING THE CORE, AND METHODS OF CALCULATING THE EFFECTS OF CORE MOVEMENT AND MATERIAL TRANSFER ON REACHING SLUMPING TEMPERATURE.

369 RELAP2 RELAP2 CALCULATES FLOW, MASS INVENTORIES, TEMPERATURES, PRESSURES, REACTIVITIES, AND TRANSIENT POWER FOR THE PRIMARY SYSTEM OF A WATER REACTOR DURING A REACTIVITY OR A LOSS-OF-COOLANT ACCIDENT. ALTHOUGH RETAINING THE SIMPLIFIED GEOMETRY (THREE VOLUMES PLUS A CORE REGION) OF THE PREVIOUS RELAP PROGRAM, MANY IMPROVEMENTS AND EXTENSIONS HAVE BEEN MADE. THE GEOMETRY CAN BE MADE TO APPROXIMATE EITHER A PRESSURIZED OR A BOILING WATER REACTOR SYSTEM. THE CORE IS TREATED AS A TWO-POINT MODEL FOR POWER GENERATION, HEAT TRANSFER, AND REACTIVITY FEEDBACKS AND AS A ONE-POINT MODEL FOR THE REACTOR KINETICS, PRESSURE BALANCES, AND FLOW BALANCES. ALSO, RELAP2 CAN BE USED FOR REACTOR SYSTEM SAFETY STUDIES INCLUDING LARGE REACTIVITY EXCURSIONS AS WELL AS THE LOSS-OF-COOLANT AND PUMP-FAILURE ACCIDENTS.

433 CONTEMPT-CONPS THE CONTEMPT-CONPS PROGRAM PREDICTS THE PRESSURE-TEMPERATURE RESPONSE OF A DRY WELL OR A DRY WELL WITH VENTS AND PRESSURE SUPPRESSION SYSTEM TO A LOSS-OF-COOLANT ACCIDENT. THE DRY WELL IS SEPARATED INTO A LIQUID REGION AND A VAPOR REGION. EACH REGION IS ASSUMED TO HAVE A UNIFORM TEMPERATURE BUT THE TEMPERATURES 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 MULTI-REGION HEAT CONDUCTION EQUATIONS. WATER AND ENERGY RATES FROM DISCHARGE OF COOLANT, BOILING OF RESIDUAL WATER BY REACTOR DECAY HEAT, SUPERHEATING OF STEAM PASSING THROUGH THE CORE, AND METAL-WATER REACTIONS ARE ASSUMED AVAILABLE FROM PREVIOUS CALCULATIONS AND ARE INPUT DATA TO THE PROGRAM. PROGRAM OUTPUT INCLUDES CONTAINMENT VOLUME PRESSURE AND TEMPERATURE, TEMPERATURES THROUGH THE BUILDING STRUCTURES, AND THE AMOUNT OF WATER, VAPOR, AND ENERGY IN THE CONTAINMENT VOLUMES. THE PRESSURE SUPPRESSION CALCULATIONS INCLUDE VENT CLEARING AND FLOW OF A TWO-COMPONENT TWO-PHASE WATER-AIR MIXTURE THROUGH THE VENTS, AND A MASS-ENERGY BALANCE IN THE WET WELL IN WHICH THE LIQUID AND VAPOR REGIONS ARE ASSUMED TO HAVE THE SAME TEMPERATURE. THERE ARE PROVISIONS FOR NORMAL BUILDING LEAKAGE, LEAKAGE FROM PENETRATIONS, A FAN COOLER SYSTEM, AND DRY AND WET WELL SPRAY SYSTEMS.

435 BURST1 BURST1 PROVIDES FOR THE STUDY OF PRESSURIZED LIQUID IN A CYLINDRICAL SYSTEM IMMEDIATELY AFTER A RUPTURE OCCURS AT ONE OR BOTH ENDS OF THE SYSTEM. PRESSURE, MASS VELOCITY, AND ENTHALPY ARE CALCULATED AT EQUALLY-SPACED MESH POINTS ALONG THE LENGTH OF THE SYSTEM. THIS IS A ONE-DIMENSIONAL REPRESENTATION, ASSUMING UNIFORM CONDITIONS THROUGHOUT ANY GIVEN CROSS-SECTIONAL AREA. FORCES ON DESIGNATED SECTIONS OF THE SYSTEM CAN ALSO BE COMPUTED, WITH PROVISION FOR DIRECTIONAL CHANGES.

443 HAA3 HAA3 EVALUATES AN APPROXIMATION TO THE GENERAL INTEGRO-DIFFERENTIAL EQUATION WHICH DESCRIBES AEROSOL BEHAVIOR. THE PHYSICAL MODEL INCLUDES PARTICLE GENERATION, BROWNIAN AND GRAVITATIONAL AGGLOMERATION, SETTLING, PLATING, AND LEAKAGE FOR SPHERICAL PARTICLES. THE APPROXIMATION IS THE REQUIREMENT THAT THE SIZE DISTRIBUTION FUNCTION BE LOG-NORMAL. THE TIME-DEPENDENT PHYSICAL QUANTITIES COMPUTED ARE (1) PLATED, SETTLED, SUSPENDED, AND LEAKED VOLUME FRACTIONS, (2) GEOMETRIC MEAN DEVIATION, NUMBER CONCENTRATION, AND GEOMETRIC MEAN RADIUS FOR THE LOG-NORMAL DISTRIBUTION OF THE SUSPENDED MATERIAL, (3) THE 50 PERCENT RADIUS (BY MASS) FOR THE SUSPENDED MATERIAL, (4) THE NUMBER AND VOLUME DISTRIBUTIONS (FUNCTIONS OF RADIUS) FOR THE SETTLED, PLATED, AND LEAKED MATERIAL, (5) THE EFFECTIVE RADII FOR THE SUSPENDED, SETTLED, AND PLATED MATERIAL.

448 FLASH4 FLASH4 IS USED TO DETERMINE THE TRANSIENT RESPONSE OF A WATER-COOLED REACTOR OR HYDRAULIC SYSTEM TO SEVERE VARIABLE PRESSURE OPERATION.

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 BLOOST5 BLOOST5 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 PRESSUR 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 SUPERHEAT 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.

267 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.

272 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. THERMAL EXPANSION OF THE FUEL AND CLADDING IS ACCOUNTED FOR.

286 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.

294 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.

299 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 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.

331 M0219(FLOT1) 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.

346 THTE THTE (TRANSIENT HEAT TRANSFER VERSION E) PROVIDES A SOLUTION CAPABILITY FOR LARGE COMPLEX, THREE-DIMENSIONAL TRANSIENT AND STEADY-STATE HEAT TRANSFER PROBLEMS WHICH CAN INCLUDE CONDUCTION, CONVECTION, AND RADIATION WITH THE OPTION TO COMPUTE FLUID FLOW RATES ON A ONE-DIMENSIONAL BASIS.

348 TOPS THE TOPS PROGRAM IS A DIGITAL SIMULATION OF PRESSURIZER DYNAMICS BASED ON A RIGOROUS APPLICATION OF THE FIRST LAW OF THERMODYNAMICS AND PHENOMOLOGICAL HEAT AND MASS TRANSFER LAWS WITH EMPIRICALLY DETERMINED COEFFICIENTS. THE PROGRAM IS USEFUL IN STUDYING THE THERMODYNAMIC PATHS OF PRESSURIZER TRANSIENTS AND IS CONVENIENT TO USE AS A DESIGN TOOL.

349 TOODEE THE TOODEE PROGRAM CALCULATES TEMPERATURES AT THE CENTER POINTS OF A TWO-DIMENSIONAL ARRAY IN X-Y, R-Z, OR R-THETA GEOMETRY. THE MESH IN THIS ARRAY MAY BE VARIABLY-SPACED. AVERAGED MATERIAL CONSTANTS ARE USED WHICH MAY BE SPATIAL AND TEMPERATURE DEPENDENT. PROVISION IS MADE IN THE PROGRAM FOR MATERIAL PHASE CHANGES. CHANNELS FOR FORCED FLOW COOLANT MAY BE INCLUDED AT EXTERIOR BOUNDARIES. IN ADDITION TO THE TEMPERATURE ARRAY, PROGRAM OUTPUT INCLUDES SURFACE TEMPERATURES AND HEAT FLUXES.

382 RAPP RAPP COMPUTES THE RELATIONSHIP AMONG MASS FLOW, PRESSURE, AND PIPING RESISTANCE (K-FACTOR) FOR HIGH VELOCITY FLOW OF A TWO-PHASE MIXTURE OF STEAM AND WATER. THE SOURCE FLUID MAY BE SUBCOOLED OR SATURATED WATER, SATURATED STEAM, OR A MIXTURE OF STEAM AND WATER. THE DOWNSTREAM PRESSURE MUST BE BELOW THE SATURATION PRESSURE OF THE SOURCE FLUID. SPECIFIC APPLICATIONS INCLUDE PRESSURIZER SURGE LINE PRESSURE DROP AND PRESSURE DISTRIBUTION DOWNSTREAM OF A RELIEF VALVE.

395 FLAC THE FLOW ANALYSIS CODE FLAC CALCULATES THE STEADY-STATE FLOW AND PRESSURE DISTRIBUTION IN AN ARBITRARY NETWORK. THE PROGRAM INCLUDES THE POSSIBILITY OF HEAT ADDITION AND MASS ADDITION IN ANY PORTION OF THE NETWORK.

396 WASP WASP USES A UNIQUE METHOD TO QUICKLY CALCULATE VARIOUS WATER AND STEAM PHYSICAL PROPERTIES OVER AN EXTREMELY WIDE RANGE.

408 TAC2D TAC2D IS DESIGNED TO TREAT TRANSIENT, TWO-DIMENSIONAL HEAT TRANSFER PROBLEMS. STEADY-STATE PROBLEMS ARE TREATED BY CONSIDERING THE PROBLEM TO BE A TRANSIENT, STARTING WITH AN ASSUMED TEMPERATURE DISTRIBUTION AND RUNNING UNTIL EQUILIBRIUM CONDITIONS ARE ESTABLISHED. GEOMETRICALLY, RECTANGULAR (X-Y), CYLINDRICAL (R,Z), OR CIRCULAR (R,THETA) COORDINATES MAY BE USED.

414 TAC3D TAC3D IS DESIGNED TO TREAT TRANSIENT, THREE-DIMENSIONAL HEAT TRANSFER PROBLEMS. STEADY-STATE PROBLEMS ARE TREATED BY CONSIDERING THE PROBLEM TO BE A TRANSIENT, STARTING WITH AN ASSUMED TEMPERATURE DISTRIBUTION AND RUNNING UNTIL EQUILIBRIUM CONDITIONS ARE ESTABLISHED. GEOMETRICALLY, THE PROBLEM MAY BE DEFINED BY EITHER RECTANGULAR (X,Y,Z) OR CYLINDRICAL (R,Z,THETA) COORDINATES.

424 GLUB1 GLUB1 SOLVES THE WATERLOGGING TRANSIENT CAUSED BY THE EXISTENCE OF A FUEL ELEMENT CLADDING DEFECT DURING AN INCREASE IN POWER. THE POWER IS EITHER - CASE 1, PUT IN AS AN ARBITRARY FUNCTION OF TIME, OR CASE 2, ASSUMED TO BE A SERIES OF LINEAR RAMPS TO A NEW STEADY-STATE WITH STEEPER AND STEEPER SLOPES. ALL THE GEOMETRY AND METAL PROPERTIES ARE INPUT QUANTITIES. WATER PROPERTIES ARE OBTAINED FROM INTERNAL TABLES. CLAD AND FUEL THERMAL, STRESS, AND STRAIN COMPUTATIONS ARE PERFORMED. THE OUTPUT CONSISTS OF INTERNAL PRESSURE AND DEFECT FLOW. FOR CASE 1, THE CLAD STRESS VERSUS TIME IS OUTPUT, AND FOR CASE 2, THE NEW STEADY-STATE POWER WHICH JUST CAUSES CLAD FAILURE IS OUTPUT.

432 COBRA2 COBRA2 CALCULATES THE STEADY-STATE FLOW, ENTHALPY AND PRESSURE DROP IN THE SUBCHANNELS OF ROD BUNDLE NUCLEAR FUEL ELEMENTS DURING BOTH NON-BOILING AND BOILING CONDITIONS. THE PROGRAM USES A MATHEMATICAL MODEL THAT INCLUDES THE EFFECTS OF TURBULENT AND DIVERSION CROSSFLOW MIXING BETWEEN THE SUBCHANNELS.

440 DYNAM DYNAM PERFORMS A DYNAMIC ANALYSIS OF ONCE-THROUGH BOILING FLOW OSCILLATIONS WITH STEAM SUPERHEAT. THE MODEL DESCRIBING THE SUPERHEAT REGIME (SINGLE-PHASE, VARIABLE DENSITY FLUID) FOR SUBCRITICAL PRESSURE OPERATION IS ALSO APPLICABLE TO THE STUDY OF ONCE-THROUGH OPERATION USING SUPERCRITICAL PRESSURE WATER.

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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.

80 SOR2 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 LOST FROM A SINGLE-PHASE FLUID IS RADIATED TO SPACE. IT ACCOUNTS FOR HEAT TRANSFER, FLUID FLOW, METEOROID PROTECTION AND RADIATOR GEOMETRY.

187 CENTRIFUGAL PUMP IMPELLER DES 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.

190 WOPEXPRT, WEIGHT OPTIMIZATION 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.

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 EXTERNA COOLING. THE FUEL ELEMENT IS COMPOSED OF THREE CONCENTRIC GRAPHIT 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-PORS SAFE-PORS 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.

266 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 COOLANT 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.

282 SEALSHELL2 (M0110) 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.

283 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.

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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.

329 M0457(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.

337 STEM STEM CALCULATES AND PUNCHES OUT PARTIALLY-COUPLED MASS, STIFFNESS AND INTERNAL LOAD FUNCTION MATRICES FOR A STRUCTURAL SYSTEM OF BEAMS HAVING PRISMATIC SEGMENTS. SHEAR DEFORMATION AND ROTATIONAL INERTIA ARE INCLUDED IN THE CALCULATIONS.

344 GEM GEM IS INTENDED PRIMARILY TO PERFORM VIBRATION STUDIES WITH THE CAPABILITY OF GENERATING INPUT FOR THE VEP (VIBRATION EIGENVALUE PROBLEM) ROUTINE AND PERFORMING ADDITIONAL OPERATIONS ON THE OUTPUT FROM THE SHO (SHOCK) SEGMENT. GIVEN A SYSTEM OF MASSES AND SPRINGS, THE VEP ROUTINE COMPUTES THE NATURAL FREQUENCIES OF THE VIBRATING SYSTEM AS WELL AS THE MODE SHAPES FOR EACH FREQUENCY. GIVEN THE MODE SHAPES, FREQUENCIES, AND MASSES OF A VIBRATING SYSTEM, THE SHO ROUTINE WILL COMPUTE THE DEFLECTIONS AND FORCES AT THE MASS POINTS.

353 SWELL2 SWELL2 WAS DEVELOPED TO PROVIDE THE FUEL-ELEMENT LIFETIME DATA REQUIRED FOR POWER PLANT OPTIMIZATION STUDIES. THE PROGRAM COMMENCES BY CALCULATING RADIAL AND AXIAL TEMPERATURES IN A FUEL ELEMENT. THE FUEL WHICH IS HOTTER THAN THE EFFECTIVE-FISSION-GAS-RELEASE-TEMPERATURE IS CONSIDERED TO RELEASE ITS FISSION GAS IMMEDIATELY TO A PLENUM, AND THE PRESSURE OF ALL PLENUM GAS IS CALCULATED. THE PRESSURE OF THE GAS RETAINED FOR A WHILE IN THE COLDER FUEL IS ALSO CALCULATED. AT EACH AXIAL POSITION THE PLENUM PRESSURE IS COMPARED WITH THE PRESSURE EXERTED BY RETAINED FISSION GAS, AND THE LARGER OF THE TWO PRESSURES IS CONSIDERED TO BE THE EFFECTIVE PRESSURE AS FAR AS CLADDING DAMAGE IS CONCERNED. THE CLADDING IS ASSUMED TO FAIL WHEN ITS CUMULATIVE DAMAGE EQUALS UNITY AT ANY AXIAL POSITION. DAMAGE COMPONENTS ARE PRIMARY CREEP, SECONDARY CREEP, RATCHETING GROWTH DUE TO THERMAL CYCLING, FATIGUE DUE TO THERMAL CYCLING, AND STRAIN DUE TO INEXORABLE FUEL SWELLING. AT EACH STEP OF THE CALCULATIONS THE FUEL ELEMENT IS SUBJECTED TO SEVERAL CONDITIONS OF ABNORMAL OPERATION (OVERPOWER, FLOW COASTDOWN, ETC.) TO ENSURE THEY COULD NOT PRODUCE ENOUGH ADDITIONAL DAMAGE TO CAUSE FAILURE AT THAT TIME.

357 SUPORAN SUPORAN SOLVES FOR STEADY-STATE DEFORMATION AND STRENGTH CHARACTERISTICS OF A NUCLEAR REACTOR CORE SUPPORT STRUCTURE. THIS STRUCTURE IS ASSUMED TO BE MADE OF TWO CIRCULAR PLATES WHICH ARE LOCATED ABOVE EACH OTHER AND ARE INTERCONNECTED PERPENDICULARLY WITH CONCENTRIC ROWS OF TUBULAR MEMBERS. THE GEOMETRY OF THE STRUCTURE AND ITS LOADING (TRANSVERSE PRESSURE AND INTERNAL TEMPERATURE GRADIENTS) ARE ALL OF AXISYMMETRICAL NATURE.

365 BOW2 BOW2 IS USED TO CALCULATE DEFLECTIONS OF CLOSELY-SPACED PARALLEL BEAMS, EACH WITH LIMITED-PIVOT SUPPORT AT ONE END, POSSIBLE BEAM INTERACTIONS AT THE OTHER END AND AT ONE INTERMEDIATE POSITION, ASSUMING AN ARBITRARY TEMPERATURE DISTRIBUTION.

378 TUBE TUBE SOLVES FOR THE STRESSES DUE TO PRESSURE AND TEMPERATURE IN A U-TUBE TYPE HEAT EXCHANGER. SPECIFICALLY, IT HANDLES A CONFIGURATION CONSISTING OF A SPHERICAL HEAD, PRIMARY TRANSITION CYLINDER, AND SECONDARY CYLINDER. THE TRANSITION CYLINDERS MAY BE CONICAL AND TAPERED IN THICKNESS, BUT THE REMAINING SHELLS ARE OF UNIFORM THICKNESS. THE SPHERICAL HEAD AND THE TRANSITION CYLINDERS MAY BE OMITTED FROM A PROBLEM.

379 SAFE-2D SAFE-2D PERFORMS THE ELASTIC STRESS ANALYSIS OF GENERAL AXISYMMETRIC, PLANE, AND COMBINED AXISYMMETRIC AND PLANE COMPOSITE STRUCTURES.

383 MO266 MO266 COMPUTES THE DYNAMIC SHOCK FORCES AND MODAL FREQUENCIES ACTING ON A LUMPED MASS, LINEAR ELASTIC MODEL OF A STRUCTURE SUBJECTED TO SHOCK SPECTRUM INPUTS. THE MODEL EMPLOYED IS A COLLECTION OF LUMPED MASSES CONNECTED BY WEIGHTLESS FLEXIBLE ELEMENTS. IF THE ORIGINAL STRUCTURE IS NOT STATICALLY DETERMINATE, REDUNDANT FORCES MUST BE INTRODUCED TO ENSURE A PRIMARY STRUCTURE THAT IS.

391 SORSDB SORSDB WAS WRITTEN TO CALCULATE THE STRESS INTENSITIES AND FATIGUE USAGE FACTORS FOR PRESSURE VESSELS (STRUCTURAL MEMBERS) IN ACCORDANCE WITH THE ASME CODE, NUCLEAR VESSELS, SECTION 3. THE INPUT REQUIRES THE BASIC MEMBRANE AND BENDING STRESSES GENERATED BY THE SOR2 CODE (ACC ABSTRACT 80) OR ANY ADAPTABLE SHELL PROGRAM. THE OUTPUT INCLUDES STRESS DIFFERENCES, STRESS INTENSITIES, AND FATIGUE USAGE FACTORS.

397 GAPL3 GAPL3 DETERMINES THE INELASTIC, LARGE-DEFLECTION BEHAVIOR OF THIN PLATES OR AXIALLY SYMMETRIC SHELLS WITH PRESSURE LOADING AND DEFLECTION RESTRAINTS.

402 SABOR4 SABOR4 IS A DISCRETE ELEMENT DISPLACEMENT PROGRAM FOR THE LINEAR-ELASTIC, STATIC, LOAD-DEFLECTION ANALYSIS OF MERIDIONALLY-CURVED, VARIABLE-THICKNESS, BRANCHED, THIN SHELLS OF REVOLUTION WHICH MAY BE SUBJECTED TO CONCENTRATED OR DISTRIBUTED EXTERNAL LOADING AND TO ISOTHERMAL OR NONUNIFORM TEMPERATURE CONDITIONS. THE PROGRAM CONSISTS OF FIVE PRIMARY SUBPROGRAMS - HAL4, MELT4, LANCE4, MAINS4, AND SUMPLOT4. THREE OF THESE (HAL4, MELT4, AND LANCE4) PROCESS EXTERNAL-MECHANICAL-LOAD OR THERMAL-LOAD DATA TO DETERMINE THE GENERALIZED NODAL FORCES FOR EACH DISCRETE ELEMENT. MAINS4 COMPUTES THE DISCRETE-ELEMENT STIFFNESS AND MASS MATRICES AND FORMS THESE MATRICES FOR THE ASSEMBLED STRUCTURE. SUMPLOT4 CALCULATES THE OTHER QUANTITIES OF INTEREST, SUMS CONTRIBUTIONS FROM VARIOUS LOADING HARMONICS, AND PREPARES RESULTS FOR CALCOMP PLOTTING.

404 FINEL FINEL IS A COLLECTION OF THREE PROGRAMS FOR THE CALCULATION OF THE ELASTIC AND PLASTIC BEHAVIOR OF STRUCTURES. THE STRUCTURES ARE BUILT UP FROM PLATES AND BARS, WHICH CAN BE ARRANGED IN EITHER 2-DIMENSIONAL OR 3-DIMENSIONAL ASSEMBLIES. THE PLATES CAN BE IN PLANE STRESS OR PLANE STRAIN. FINEL2 HANDLES 2-D STRUCTURES ONLY. FINEL3 CAN HANDLE BOTH 2- AND 3-D STRUCTURES, BUT IS SLOWER.

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412 MANE1 THE MANE1 PROGRAM FINDS THE MAGNETIC FLUX IN EACH BRANCH OF A MAGNETIC NETWORK CONSISTING OF A NUMBER OF BRANCHES OF IRON AND AIR. THE PROBLEM IS ONE STEP IN THE DESIGN OF SLOW SPEED RELUCTANCE MOTORS IN WHICH THE MAGNETIC FLUX IS DETERMINED IN ALL COMPONENT PARTS FOR A POSITION OF THE ROTOR WITH RESPECT TO THE STATOR.

449 CYGRO3 CYGRO3 IS AN EXTENSIVE MODIFICATION OF CYGRO1 AND CYGRO2 (ACC ABSTRACT 266). BASICALLY THE PROGRAM CALCULATES TEMPERATURES, DEFORMATION, AND STRESSES IN CLADDED FUEL RODS AS A FUNCTION OF A HISTORY OF POWER AND COOLANT CONDITIONS. AXIAL AND CIRCUMFERENTIAL UNIFORMITY ARE ASSUMED. BUBBLE GROWTH AND MIGRATION ARE INCLUDED. THE MAIN CHANGES FROM CYGRO1 AND CYGRO2 ARE IN THE AREA OF VOID MIGRATION, FUEL CRACKING, CLAD COLLAPSE, REPRESENTATION OF IN-PILE CREEP AND CLAD ANISOTROPY.

45 GRACE1 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.

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 FARSE-1A 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 SHOE 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 CYLINDRICAL CAVITY.

343 M0756(LETO) LETO WILL SOLVE THE GAMMA RAY TRANSPORT AND ENERGY DEPOSITION PROBLEM IN ONE-DIMENSIONAL LAMINAR SLAB GEOMETRY. THE ENERGY GROUP SCHEME IS EMPLOYED TO ACCOUNT FOR PHOTON ENERGY DEGRADATION. AN ARBITRARY EXTERNAL SPATIAL ISOTROPIC SOURCE MAY BE SPECIFIED WITH AN ARBITRARY ENERGY SPECTRUM. THE BOUNDARY CONDITIONS MAY BE (A) FREE BOUNDARIES WITH ARBITRARY INCIDENT, (B) SYMMETRY ON THE LEFT ARBITRARY INCIDENT ON THE RIGHT, AND (C) SYMMETRY ON BOTH ENDS.

429 ASPIS ASPIS COMPUTES THE ENERGY, DOSE AND ENERGY DEPOSITION BUILDUP FACTORS FOR MONOENERGETIC GAMMA RAYS FOR A PLANE ISOTROPIC, PLANE MONODIRECTIONAL, OR PLANE SLANT SOURCE, IN AN ARBITRARY LAMINAR ARRAY.

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.

387 CITATION CITATION IS DESIGNED TO ATTACK THE REACTOR CORE DEPLETION AND DYNAMICS PROBLEMS IN A GENERAL SENSE. A VARIETY OF MATHEMATICAL FORMULATIONS IS AVAILABLE TO CHOOSE FROM - FOR EXAMPLE, THE GENERAL NEUTRON-FLUX-EIGENVALUE PROBLEM MAY BE SOLVED IN ONE-, TWO- OR THREE-DIMENSIONAL GEOMETRY WITH ARBITRARY SCATTERING IN THE DISCRETE-ENERGY, FINITE-DIFFERENCE DIFFUSION-THEORY FORMULATION. CERTAIN OTHER FORMULATIONS ARE BEING ADDED AND STILL OTHERS WILL BE INCORPORATED. NOW THE PROGRAM WILL SOLVE EIGENVALUE PROBLEMS WITHIN A FINITE-DIFFERENCE REPRESENTATION OF THE DIFFUSION THEORY APPROXIMATION TO NEUTRON TRANSPORT IN MULTIGROUP FORM WITH ARBITRARY SCATTERING. THE EIGENVALUE PROBLEM MAY INVOLVE THE DETERMINATION OF THE EFFECTIVE MULTIPLICATION FACTOR, OR BE A DIRECT CRITICALITY SEARCH ON BUCKLING, ON 1/V ABSORBER, OR ON THE CONCENTRATIONS OF SPECIFIED NUCLIDES WHICH MAY INCLUDE FUEL. THE ADJOINT PROBLEM MAY BE SOLVED AND PERTURBATION CALCULATIONS PERFORMED. UP TO THREE GEOMETRIC DIMENSIONS MAY BE CONSIDERED INCLUDING X-Y-Z, R-THETA-Z, AND HEXAGONAL-Z COORDINATE SYSTEMS.

189 SNC CALCULATION OF SN CONSTANTS FOR DSN AND TDC.

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 P1 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*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 CCNSISTS OF TRIANGULAR ELEMENTS AND NODES, UNIAXIAL ELEMENTS, SUCH AS REINFORCEMENT BARS, TENCONS, AND ANCHORS, AND BIAXIAL MEMBRANES, SUCH AS ANY THIN SHELL OR LINER. THE PUNCHED OUTPUT SERVE AS DIRECT INPUT DATA TO THE SAFE CODES.

345 GAND GAND PREPARES THE CROSS SECTICNS NEEDED FCR DETAILED COMPUTATIONS OF NEUTRCN ENERGY SPECTRA IN FAST REACTORS FROM A FILE OF BASIC NUCLEAR DATA IN THE ENDF/B FORMAT.

373 BL47 BL47 IS A PLOTTING ROUTINE DESIGNED FOR PLANE STRUCTURES THAT ARE TO UNDERGO STRESS ANALYSIS. POINTS AND LINES ARE INPUT IN VARICUS PARAMETRIC FORMS, AND CURVED SEGMENTS ARE DRAWN BETWEEN GIVEN POINTS ALONG THE GIVEN LINES. THE PROGRAM MAY BE USED AS A DRAFTING TOOL TO CONSTRUCT ENGINEERING DRAWINGS. BL47 USES THREE PCINTS ON A STRAIGHT LINE SEGMENT TO CBTAIN DIMENSICNS FOR SEALSHELL2 (ACC ABSTRACT 282) INPUT DATA.

406 DATATRAN 2-D GEOMETRY INPUT THIS SERIES OF DATATRAN MCDULES IS DESIGNED TO FACILITATE THE PREPARATION OF GECMETRIC INPUT TO 2-DIMENSICNAL FINITE DIFFERENCE PROGRAMS FROM DATA READILY TRANSCRIBEC FROM ENGINEERING DRAWINGS AND TO CHECK AND EDIT THE INPUT DATA GENERATEC.

159 MOMUS 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.

350 ETOE ETOE (ENDF/B TO MC**2 DATA CONVERSION) ACCEPTS CROSS SECTION DATA FROM A MODE 2 ENDF/B TAPE (SEE REFERENCE 3) AND PREPARES THE BINARY CROSS SECTION AND LEGENDRE POLYNOMIAL TAPE FOR THE MC**2 CODE WRITTEN BY ARGONNE NATIONAL LABORATORY.

351 ECSIL ECSIL (EXPERIMENTAL CROSS SECTION INFORMATION LIBRARY) IS A SYSTEM FOR THE STORAGE, RETRIEVAL, AND DISPLAY OF EXPERIMENTAL NEUTRON DATA. THE HEART OF THE ENTIRE SYSTEM IS THE COLLECTION OF DESIGNATORS USED TO IDENTIFY THE TYPE OF NEUTRON DATA, E.G., WHETHER THE MEASUREMENT IS A FISSION CROSS SECTION, AN ANGULAR DISTRIBUTION FOR EMERGENT NON-ELASTIC NEUTRONS BETWEEN TWO NEUTRON ENERGIES, ETC. THERE ARE THREE DICTIONARIES USED FOR INPUT TO THE DATA FILE - ONE FOR THE REACTION-TYPE DESIGNATOR, ONE FOR THE STATUS OF THE DATA, AND ONE TO FLAG THE PROGRAM TO PERFORM CERTAIN CONVERSIONS. EXPERIMENTAL NEUTRON DATA ARE, IN GENERAL, COMPOSED OF TWO INTERDEPENDENT, BUT LOGICALLY SEPARABLE PARTS, THE BIBLIOGRAPHIC INFORMATION AND THE ACTUAL EXPERIMENTAL VALUES. THE REFERENCE ACCESSION NUMBER, WHICH IS ASSIGNED TO A REFERENCE AND ITS ASSOCIATED SET OF DATA AS THEY ARE ACQUIRED, SERVES AS A LINK BETWEEN THE BIBLIOGRAPHIC AND THE EXPERIMENTAL DATA FILES. AFTER A REFERENCE ACCESSION NUMBER IS ASSIGNED TO A NEW REFERENCE, THE FOLLOWING INFORMATION IS ENTERED INTO THE BIBLIOGRAPHIC FILE - THE COMPLETE BIBLIOGRAPHIC CITATION, THE LABORATORY WHERE THE MEASUREMENT WAS PERFORMED, A BRIEF DESCRIPTION OF THE EXPERIMENTAL TECHNIQUE, CORRECTIONS THAT HAVE BEEN MADE TO THE RESULTS, AND NORMALIZATIONS, IF ANY. IN ADDITION, ANY CHANGES MADE TO THE DATA (RENORMALIZATIONS TO BETTER STANDARDS, CORRECTIONS, ETC.) ARE RECORDED HERE. SUPPLEMENTARY REFERENCES ARE CARRIED ALONG AS SEE ALSOS.

375 SCORE2 SCORE IS AN INTERACTIVE NEUTRON CROSS SECTION EVALUATION SYSTEM.

384 CHECKER/CRECT/CAMMET/PLOTFB/ THIS PACKAGE OF FIVE PROGRAMS IS DESIGNED FOR PROCESSING ENDF/B (EVALUATED NUCLEAR DATA FILE VERSION B) TAPES.

CHECKER CHECKS THAT THE ENDF/B TAPES ARE IN PROPER FORMAT AND ALL FIELDS ARE WITHIN SPECIFIED LIMITS, RATHER THAN THE PHYSICS OF THE DATA LIBRARY. ANGULAR DISTRIBUTIONS RECONSTRUCTED FROM LEGENDRE COEFFICIENTS ARE EVERYWHERE POSITIVE.

CRECT PROVIDES A MEANS OF CORRECTING ASSEMBLED DATA ON A TAPE BY INSERTION AND DELETION OF DATA.

CAMMET SELECTIVELY MERGES DATA FROM ONE OR TWO ENDF/B LIBRARY TAPES ONTO A FINAL TAPE. THE MODE (BCD OR BINARY) AND ARRANGEMENT (STANDARD OR ALTERNATE) MAY BE CHANGED DURING THIS PROCESS.

PLOTFB PROCESSES ENDF/B LIBRARY TAPES WHICH CONTAIN DATA EMBEDDED WITHIN A NECESSARY LIBRARY STRUCTURE IN ORDER TO PRODUCE COMPREHENSIVE LISTINGS AND/OR PLOTS. THE LISTINGS AND/OR PLOTS CONTAIN AN EXTENSIVE AMOUNT OF INFORMATION RELATED TO THE DATA, SUCH AS TEMPERATURE DEPENDENCE, PHYSICAL UNITS OF THE DATA, INTERPOLATION LAWS FOR THE DATA, CRYPTIC TITLES DEFINING THE REACTION TYPE, ETC.

SLAVE3 PROVIDES MODULAR SUBROUTINES WHICH CAN BE ASSEMBLED TO RETRIEVE AND PROCESS ENDF/B DATA FOR A SPECIFIC PROBLEM.

386 DATATRAN DATATRAN SUPPLIES A LINKAGE SYSTEM FOR MODULARIZED PROGRAMS, ENABLES A HIERARCHICAL NAMING TECHNIQUE, AND SIMPLIFIES HANDLING OF STRUCTURED DATA LISTS.

403 TIGIR2 TIGIR IS A MODULAR PROGRAM DESIGNED TO GENERATE AND MAINTAIN LIBRARIES OF DOCUMENT INFORMATION (ABSTRACTS, BIBLIOGRAPHIC INFORMATION, INDEX TERMS, AND OTHER SIGNIFICANT INFORMATION), TO RETRIEVE THE INFORMATION SELECTIVELY, TO PRINT THE RETRIEVED INFORMATION IN VARIOUS OUTPUT FORMATS, AND TO GENERATE STATISTICS OF THE INFORMATION FILES.

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/CROSSPLOT DATA TAPE AUTOMATIC PLOTS ARE GENERATED 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.

409 LARCA MULTIGROUP CROSS SECTIONS ARE WEIGHTED BY THE FLUX IN THE APPROPRIATE GROUPS AND REGIONS, AND CROSS SECTIONS FOR THE ASSEMBLY ARE COMPUTED AND PUNCHED IN DTF4 FORMAT. THE PROGRAM ALSO COMPUTES THE INFINITE MEDIUM FLUX, REACTION RATES, AND THE INFINITE MULTIPLICATION FACTOR AND MATERIAL BUCKLING.

434 HEATMESH HEATMESH IS USED TO GENERATE GEOMETRICAL DATA REQUIRED FOR STUDIES OF HEAT TRANSFER IN AXISYMMETRIC STRUCTURES REPRESENTED AS SURFACES OF REVOLUTION. THE PROGRAM CONSISTS OF TWO DISTINCT PHASES. THE FIRST SUBDIVIDES THE GIVEN PARTS INTO A NODAL NETWORK AND EVALUATES THE GEOMETRICAL PROPERTIES OF THE NODES. THE SECOND DETERMINES ADJACENT NODES AND EDITS GEOMETRICAL DATA FOR THE THERMAL MODEL.

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. PHOTOFRACTION 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. PHOTOFRACTION 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 COINC (COMPUTER CODE FOR REDU 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*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*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 TOAD IS USED TO PROCESS AND ANALYZE GAMMA RAY SPECTRA.

390 CORGAM A CORRELATION ALGORITHM IS CODED TO ALLOW THE UNFOLDING OF COMPLEX GAMMA-RAY SPECTRA TYPICALLY COLLECTED IN A NEUTRON ACTIVATION ANALYSIS PROCEDURE. CORGAM (1) WILL COMPENSATE FOR ELECTRONIC SHIFTS IN THE DATA, (2) WILL CORRECT FOR BACKGROUND, (3) WILL NORMALIZE THE DATA TO A FIXED NEUTRON FLUX LEVEL, (4) ALLOWS A CHOICE OF WEIGHTING FACTORS, AND (5) ALLOWS A CHOICE OF METHODS FOR CALCULATION OF STANDARD DEVIATIONS. THE CODE REQUIRES A MATRIX OF REFERENCE GAMMA-RAY SPECTRA. THESE SPECTRA CAN BE IN A RAW-DATA FORM. ALL OF THE MODIFICATIONS AVAILABLE TO THE COMPLEX GAMMA-RAY SPECTRA ARE AVAILABLE TO THE REFERENCE GAMMA-RAY SPECTRA. IN ADDITION, A DECAY CORRECTION IS AVAILABLE FOR THE REFERENCE GAMMA-RAY SPECTRA. ONLY THE REFERENCE GAMMA-RAY SPECTRA THAT HAVE INTENSITY COEFFICIENTS WHICH ARE SIGNIFICANT AT A PRESCRIBED LEVEL OF SIGNIFICANCE ARE RETAINED IN THE FINAL SOLUTION. THE INTERMEDIATE SOLUTIONS, I.E., THOSE SOLUTIONS THAT CONTAIN REFERENCE GAMMA-RAY SPECTRA WHICH HAVE NONSIGNIFICANT INTENSITY COEFFICIENTS AT THE PRESCRIBED LEVEL, ARE PRINTED OUT. THEREFORE, SEVERAL SOLUTIONS ARE IMBEDDED IN THE FINAL SOLUTION.

394 JITER JITER COMPUTES THE FOLLOWING QUANTITIES MEASURED IN REACTOR FLUCTUATION EXPERIMENTS - THE DISPERSION PARAMETER Y, THE MODIFIED COEFFICIENT OF CORRELATION MCC, THE FREQUENCY DEPENDENT PART OF THE POWER SPECTRAL DENSITY PSD, AND THE CROSS POWER SPECTRAL DENSITY CPSD.

413 ALPHA-M ALPHA-M IS USED FOR DETERMINING RADIOISOTOPES BY LEAST SQUARES RESOLUTION OF THE GAMMA RAY SPECTRA. IT CAN HANDLE A VERY LARGE LIBRARY OF GAMMA RAY SPECTRA AND TAKES INTO ACCOUNT CORRECTIONS SUCH AS BACKGROUND SUBTRACTION, COUNTING TIME, DECAY TIME, DEAD TIME, AUTOMATIC COMPENSATION FOR GAIN AND THRESHOLD SHIFTS, SIZE OF THE ALIQUOT, VOLUME REDUCTION PRIOR TO COUNTING, AND SO ON.

421 MICHRD THE VICKERS PYRAMID NUMBER AND THE MICROHARDNESS INCREMENT FRACTION ARE COMPUTED FROM FILAR MICROMETER EYEPIECE READINGS. THESE READINGS ARE MADE ON INDENTS, WHICH HAVE BEEN MADE BY AN INDENTER, IN THE MATERIAL BEING TESTED.

425 REDUX REDUX IS USED FOR THE REDUCTION OF DATA FROM REACTOR FLUCTUATION EXPERIMENTS. FROM INPUT COUNT SAMPLES RECORDED BY ONE OR TWO COUNTING CHANNELS, THE PROGRAM COMPUTES UNBIASED SAMPLE ESTIMATES OF MOMENTS AND FUNCTIONS OF MOMENTS OF THE COUNT DISTRIBUTIONS, INCLUDING THE DISPERSION PARAMETER Y AND THE MODIFIED COEFFICIENT OF CORRELATION MCC. STANDARD ERROR ESTIMATES ARE COMPUTED FOR Y AND MCC.

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 = SUMMATION OVER I OF A(I)*(X**(I))
(2) EXPONENTIAL Y = A*EXP(BX)
(3) COSINE Y = A*COS(B(X+C))
(4) SERIES OF CUBICS Y = A(J)+B(J)*X+C(J)*(X**2)+D(J)*(X**3)
(5) FOURIER SERIES

62 LOS 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) = SUMMATION OVER J OF A(J) * EXP(-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)).

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.

327 DAFT1 DAFT1 IS A PROGRAM FOR WEIGHTED LEAST SQUARES FITTING OF 0.0253 EV NEUTRON DATA FOR FISSILE 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.

364 SNEQ SNEQ CONSISTS OF THE TWO CODES, SNAP AND EQPLT, WHICH HAVE BEEN MERGED DUE TO THEIR COMMON USE OF THE SLIP COMPILER. SNAP INTERPRETS AND SOLVES PSEUDO-FORTRAN INPUT EQUATIONS REPRESENTING NONLINEAR ALGEBRAIC SYSTEMS. EQPLT INTERPRETS PSEUDO-FORTRAN INPUT EQUATIONS AND CALCULATES AND PLOTS MULTIPLE CURVES ON A SINGLE GRAPH. EQPLT IS USEFUL FOR PARAMETER STUDIES.

407 DATATRAN UTILITY MODULES PLOT2 WAS DESIGNED TO PROVIDE ROUTINE X-Y PLOTS USING THE CALCOMP PLOTTER, MICROFILM UNIT, OR PRINTER AS THE OUTPUT DEVICE. PLOT3 WAS DESIGNED TO PREPARE CONTOUR, PERSPECTIVE, AND STEREOGRAPHIC PLOTS FOR THE CALCOMP PLOTTER OR MICROFILM UNIT OF FUNCTIONS OF TWO VARIABLES.

411 M0661, M0657, M0626 M0661, M0657, AND M0626 PERFORM STATISTICAL ANALYSES OF DATA BASED ON A LEAST SQUARES POLYNOMIAL FIT.

428 DOGGY DOGGY CAN PERFORM MOST ROUTINE FORM SHEET CALCULATIONS. THE PROGRAM CAN HANDLE COMMON ARITHMETIC MANIPULATIONS ON COLUMNS OF INPUT SUCH AS ADDITION, SUBTRACTION, MULTIPLICATION, AND DIVISION. IT ALSO HAS PROVISIONS FOR THE USE OF SPECIAL FUNCTIONS SUCH AS LOGARITHMIC, TRIGONOMETRIC, ARC TRIGONOMETRIC, HYPERBOLIC, EXPONENTIAL, SQUARE ROOT, MAXIMA, MINIMA, AND RAISING A NUMBER TO ANY POWER. DOGGY ALSO CALCULATES WATER PROPERTIES SUCH AS THERMAL CONDUCTIVITY, VISCOSITY, AND PRANDTL NUMBER OVER A WIDE RANGE OF TEMPERATURE AND PRESSURE.

442 SIMPLE1 SIMPLE1 COMPILES AND EXECUTES MULTISTATEMENT CALCULATIONS TYPED IN A FAMILIAR ALGEBRAIC NOTATION AT A TIME-SHARED TERMINAL, IMMEDIATELY RETURNING REQUESTED RESULTS TO THE TERMINAL AND PERMITTING AD LIB CONTINUATION OF A CALCULATION. THE USER MAY USE THE SYSTEM AS AN EXTENDED AND POWERFUL ELECTRONIC CALCULATOR OR AS A CALCULATOR OF SMALL LOAD-AND-GO RUNS. ERRORS ARE DETECTED AND CORRECTED ON AN AS-YOU-GO BASIS. CALCULATIONS MAY INCLUDE INTERACTIVE INPUT/OUTPUT, FUNCTIONS, AND LOOPS.

444 ROPE ROPE IS USED TO CALCULATE ROOTS OF POLYNOMIALS.

445 LIZARD4 LIZARD4 IS USED TO SOLVE NON-LINEAR, ORDINARY DIFFERENTIAL EQUATIONS AS A ONE-SHOT EFFORT. IT WAS WRITTEN TO SOLVE INITIAL VALUE EQUATIONS, THAT IS, THE VALUES OF THE DEPENDENT VARIABLES MUST BE SPECIFIED AT SOME INITIAL VALUE OF THE INDEPENDENT VARIABLE. ONE-DIMENSIONAL STEADY-STATE BOUNDARY VALUE PROBLEMS CAN BE SOLVED WITH LIZARD4, PROVIDING AN ITERATIVE METHOD IS EMPLOYED BY THE USER WHERE ONE OF THE BOUNDARY CONDITIONS IS SPECIFIED AND INTEGRATION PROCEEDS UNTIL THE ALTERNATE BOUNDARY CONDITION IS MET.

446 MOST MOST IS DESIGNED TO VARY A SET OF COORDINATES (X1,X2, . . . ,XN) REPRESENTING THE VECTOR X IN SUCH A WAY THAT A SPECIFIED FUNCTION Y(X), Y(X) GREATER THAN OR EQUAL TO 0, IS MINIMIZED.

419 CASCADE/CLUSTER CASCADE IS A SIMULATION OF THE COLLISION CASCADE RESULTING FROM THE DISPLACEMENT OF A PRIMARY KNOCK-ON ATOM BY AN ENERGETIC NEUTRON IN A BODY-CENTERED-CUBIC OR FACE-CENTERED-CUBIC CRYSTAL STRUCTURE. CLUSTER ANALYZES THE SPATIAL DISTRIBUTION OF THE RESULTING RADIATION DAMAGE IN TERMS OF VACANCY AND INTERSTITIAL CLUSTERS, TAKING INTO ACCOUNT THE EXACT CRYSTAL STRUCTURE.

422 SPECTRA SPECTRA COMPUTES THE NUMBER AND SPECTRUM OF PRIMARY KNOCK-ON ATOMS RESULTING FROM NEUTRON COLLISION SEQUENCES IN A MATERIAL UNDER IRRADIATION.

356 GAF/GAR DATA TAPES USED IN GG A BENCHMARK STUDY OF ZPR-III ASSEMBLY 48 USING ENDF/B CROSS SECTIONS WAS UNDERTAKEN TO IDENTIFY POSSIBLE CROSS SECTION DISCREPANCIES IN THE MICROSCOPIC ENDF/B DATA. THIS WORK WAS DONE FOR THE CSEWG TESTING SUBCOMMITTEE AS PART OF THEIR PHASE I DATA TESTING. THIS PACKAGE CONTAINS THE CROSS SECTION DATA GENERATED FOR THIS STUDY IN THE FORM OF THE ULTRA-FINE GROUP CROSS SECTIONS OF THE MATERIALS OF ZPR-III ASSEMBLY 48 IN THE FORMAT OF THE GGA GAF/GAR PROGRAM DATA TAPES.

447 ETOG1 DATA FOR MUFT AND GAM THE DATA REQUIRED FOR THE CREATION OF MUFT4, MUFT5, GAM1 AND GAM2 LIBRARIES WERE GENERATED FROM THE BROOKHAVEN NATIONAL NEUTRON CROSS SECTION CENTER ENDF/B TAPES 114 THROUGH 117 BY THE ETOG1 PROGRAM (ACC ABSTRACT 437).

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ANL  3600 F36   RS P  T B$ MC**2, ENDF MULTIGROUP X-SECTION CALCULATION      355
ANL  3600 F36   RS P  T C$ SNARG-1D, 1-D MULTI-GP DISCRETE ORDINATE CALC     288
ANL  3600 F36   RS P  T I$ SUPORAN, REACTOR CORE SUPPORT STRESS ANALYSIS     357
ANL  3600 F36   RS P XT C$ MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE     262
ANL  3600 F36   RSBP    A$ MISH-MASH, RESONANCE INTEGRAL CALC HOMOGENEOUS    214
ANL  3600 F36   RSBP    G$ CHEMLOC2, CORE HEATING CLADDING-STEAM REACTION    366
ANL  3600 F36   RSBP    I$ SWELL2, FUEL ELEMENT LIFETIME ANALYSIS            353
ANL  3600 F36   RSBP XT B$ RABBLE,WLIB,FLAT, RESONANCE ABSORPTION, CELL      281
ANL  3600 F36   RSBP XT B$ RIFF-RAFF, RESONANCE INTEGRAL CALC 2-REG CELL     213
ANL  3600 F4    RSBP  T C$ 2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION       325
ANL  3600 F63   RS  LXT B$ GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC         33
ANL  3600 F63   RSBP    A$ NEARREX, COMPOUND NUCLEUS X-SECTION CALC          171
ANL  3600 F63   RSBP    E$ R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS      255
ANL  3600 F63   RSBP    E$ R102, SPACE-INDEPENDENT INVERSE KINETICS CALC     168
ANL  3600 F63   RSBP    C$ COINC, COINCIDENCE COUNTING DATA REDUCTION        248
ANL  3600 F63   RSBP  T A$ 2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS     254
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ANL  3600 F63   RSBP  T E$ TRAFICORPORATION, TRANSFER FUNCTION SYNTHESIS     135
ANL  3600 F63   RSBP  T G$ R153, PARAMETRIC SITE REQUIREMENT STUDY           172
ANL  3600 F63   RSBP  T H$ ARGUS, TRANSIENT TEMPERATURE CALC CYLINDER        152
ANL  3600 F63   RSBP  T M$ XLIBIT, X-SECTION LIBRARY UTILITY ROUTINE         181
ANL  3600 F63   RSBP XT B$ THERMOS(ANL), THERMAL SPECTRUM X-SECTION CALC     184
ANL  3600 F63   RSBPL T J$ MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY       143
ANL  6600 F4    RS P  T F$ SAS1A, FAST REACTOR POWER AND FLOW TRANSIENTS     400
ANL), THERMAL SPECTRUM X-SECTION CALC  ANL  3600 F63   RSBP XT B$ THERMOS(   184
ANNULAR VOID X-SECTION CALCULATION     GEC   625 F4    RS P    B$ AVOID,     276
APDA 3600 ASAF4 RS P  T M$ ETOE, ENDF/B TO MC**2 DATA CONVERSION             350
APDA 7094 F2    RS P    F$ WEAK EXPLOSION, COUPLED NEUTRON-HYDRODYNAMICS     145
APDA 7094 F4    RS P    F$ MARS, 2-D EXCURSION CALCULATION R-Z GEOMETRY      293
APD  635 F4     RS P    C$ RAPFU, FUEL CYCLE PARAMETERS FAST BREEDERS        372
ARES2, RESONANCE INTEGRAL X-SECTION CALC       AI   7090 F2    RS PL T A $    89
ARES2, RESONANCE INTEGRAL X-SECTION CALC       CDC  1604 F63   RS PL T A $    89
ARGUS, TRANSIENT TEMPERATURE CALC CYLINDER     ANL  3600 F63   RSBP  T H $   152
ASAF4 RS P  T M$ ETOE, ENDF/B TO MC**2 DATA CONVERSION          APDA 3600     350
ASPIS, GAMMA RAY SOURCE BUILDUP FACTOR CALC    BAPL 6600 F+ASC RS P XT J $ R429
ASSAULT, 2-D MULTI-GP DIFFUSION DEPLETION CODE ORNL 7090 F+FAP RS P  T D $   240
ASSEMBLER FOR FLOCO2 INSTRUCTION SET      PW   1604 F+CDP RS P  T P$ LAG,    186
ASSEMBLY 48 GAFGAR ENDF/B DATA TAPES   GGA  1108 BIN   R   L T Z$ ZPR-III   356
ATHENA4, INELASTIC SCATTERING FORM FACTORS     ORNL  360 F4    RS P  T A $   417
ATTENUATION CYL SPHERE GEOM  AI   7090 F2    RS P    J$ GRACE2, GAMMA-RAY     46
ATTENUATION CYL SPHERE GEOM  CDC  1604 F63   RS P    J$ GRACE2, GAMMA-RAY     46
ATTENUATION SLAB GEOMETRY    AI   7090 F2    RS P    J$ GRACE1, GAMMA-RAY     45
ATTENUATION SLAB GEOMETRY    CDC  1604 F63   RS P    J$ GRACE1, GAMMA-RAY     45
AVERAGE X-SEC CALC  BNL  6600 F4    RS P    A$ AVERAGE, UNRESOLVED REGION   376
AVERAGE X-SEC CALC  BNL  7094 F4    RS P    A$ AVERAGE, UNRESOLVED REGION   376
AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC  BNL  6600 F4    RS P    A $   376
AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC  BNL  7094 F4    RS P    A $   376
AVERAGING      GGA  7044 F4    RS P    L$ GAVER, ENERGY INTERVAL X-SECTION  218
AVOID, ANNULAR VOID X-SECTION CALCULATION      GEC   625 F4    RS P    B $   276
AX-TNT, COUPLED NEUTRONICS-HYDRODYNAMICS SPH   PW   1604 F63   RS P    F $   191
AXFLU, HEAT TRANSFER MOLTEN FUEL TUBE BUNDLES  LASL 7094 F2    RSBP   H $   182
AXISYM LOAD  BAPL 6600 F4    RS P XT I$ SEALSHELL2, SHELL STRESS ANALYSIS R282
AXISYMMETRIC LOAD   GGA  7044 F4    RS P  T I$ SAFE-PCRS, STRESS ANALYSIS   250
AXISYMMETRIC LOAD GGA  7044 F4    RS P  T I$ SAFE-AXISYM, STRESS ANALYSIS   251
AXISYMMETRIC STRESS ANALYSIS  GGA  1108 F4    RS P    I$ SAFE-2D, PLANE +  379
AXTHRM, HEAT TRANSFER SOLID FUEL TUBE BUNDLES  LASL 7094 F2    RSBP   H $   183
AX1, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE   CDC  3600 F63   RSBP   F $   102
B&W  6600 F+CCM RS P XT C$ FARED, 1-D FAST REACTOR DESIGN & SURVEY STUDY    427
B-W MULTI-LEVEL CONVOLUTION    PPCO 7040 F4    RS P    A$ COMBCO, COMBINED  203
B-W RESONANCE DATA     GGA  7044 F4    RS P    A$ FASDOP, X-SECTIONS FROM   216
B-W RESONANCE PARAMETERS   WANL 7094 F2    RS P    A$ EXT, X-SECTIONS FROM  238
B-W X-SEC CALC     BNL  6600 F4    RS P    A$ SIGPLOT, RESOLVED MULTILEVEL 377
B-W X-SEC CALC     BNL  7094 F4    RS P    A$ SIGPLOT, RESOLVED MULTILEVEL 377
BAM, S4 CYL CELL CODE AND TEMPEST COMBINATION  AI   7090 F2    RS P    B $   108
BAPL 6600 F+ASC RS P XT C$ PDQ7, 1,2 CR 3-D FEW-GP DIFFUSION DEPLETION      R275
BAPL 6600 F+ASC RS P XT J$ ASPIS, GAMMA RAY SOURCE BUILDUP FACTOR CALC      R429
BAPL 6600 F+CCM RS P XT I$ GAPL3, INELASTIC LARGE DEFLECTION STRESS STUDY   R397
BAPL 6600 F4    RS   XT B$ MO807, 2-D DIFFUSION ABSORPTION REMOVAL X-SECS   R280
BAPL 6600 F4    RS   XT C$ MO648, 1-D SLAB TRANSPORT WITH SLOWING DOWN      R342
BAPL 6600 F4    RS   XT H$ WATER, STEAM TABLES 14.5-14,500PSIA 32-472DEGF   R267
BAPL 6600 F4    RS   XT I$ CYGRO2, STRESS ANALYSIS CYL FUEL ELEMENT         R266
BAPL 6600 F4    RS P    C$ CINDER,MO102, POINT DEPLETION FISSION PRODUCT     313
BAPL 6600 F4    RS P  T O$ JITER, FLUCTUATION EXPERIMENT ANALYSIS           R394
BAPL 6600 F4    RS P XT A$ PUN1, UNRESOLVED RESONANCE INTEGRALS X-SECS      R359
BAPL 6600 F4    RS P XT A$ SUMOR, S-WAVE NEUTRON X-SECTION CALCULATION      R399
BAPL 6600 F4    RS P XT E$ PAXO2, HARMONY-PDQ X-SECTION GENERATION CODE     R426
BAPL 6600 F4    RS P XT B$ RESQ2,RESQO,DBF1, RESONANCE INTEGRAL HEX CELL    R285
BAPL 6600 F4    RS P XT C$ BE21, FEW-GP DISCRETE ORDINATES SLAB GEOMETRY    R398
BAPL 6600 F4    RS P XT F$ NOWIG, 1-D 2-GP KINETICS TEMPERATURE FEEDBACK    R371
BAPL 6600 F4    RS P XT F$ TWIGL, 2-D 2-GP SPACE-TIME DIFFUSION FEEDBACK    R338
BAPL 6600 F4    RS P XT F$ WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM      R274
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BAPL 6600 F4     RS P XT G$ FLASH3, LOSS-OF-COOLANT ACCIDENT ANALYSIS          R295
BAPL 6600 F4     RS P XT G$ FLASH4, FULLY-IMPLICIT TRANSIENT SIMULATION        R448
BAPL 6600 F4     RS P XT G$ M0555,ACT1, LOSS-OF-COOLANT ACCIDENT ANALYSIS      R284
BAPL 6600 F4     RS P XT H$ FIGRO, LSBR FUEL SWELLING TEMPERATURE STUDY        R272
BAPL 6600 F4     RS P XT H$ FLOT1,M0219, PWR FLOW TRANSIENT ANALYSIS           R331
BAPL 6600 F4     RS P XT H$ GLUB1, WATER-LOGGED FUEL ELEMENT ANALYSIS          R424
BAPL 6600 F4     RS P XT H$ HOT2, 2-D TRANSIENT HEAT CONDUCTION PROGRAM        R286
BAPL 6600 F4     RS P XT H$ M0899,HOH, STEAM TABLES 14.5-2538 PSIA             R294
BAPL 6600 F4     RS P XT H$ TOPS, TRANSIENT THERMODYNAMICS OF PRESSURIZERS     R348
BAPL 6600 F4     RS P XT H$ WASP, WATER AND STEAM THERMODYNAMIC PROPERTIES     R396
BAPL 6600 F4     RS P XT I$ CYGRO3, OXIDE FUEL ROD STRESS & DEFORMATION        R449
BAPL 6600 F4     RS P XT I$ MANE1, RECTANGULAR MAGNETIC NETWORK SOLUTION       R412
BAPL 6600 F4     RS P XT I$ M0266, LINEAR ELASTIC STRUCTURAL DYNAMICS          R383
BAPL 6600 F4     RS P XT I$ M0457,PIPE, ELASTIC STRESS OF PIPING SYSTEM        R329
BAPL 6600 F4     RS P XT I$ M0552, DYNAMIC ANALYSIS LINEAR ELASTIC SYSTEMS     R283
BAPL 6600 F4     RS P XT I$ SEALSHELL2, SHELL STRESS ANALYSIS AXISYM LOAD      R282
BAPL 6600 F4     RS P XT J$ M0756,LETO, 1-D SLAB GAMMA-RAY TRANSPORT           R343
BAPL 6600 F4     RS P XT L$ BL47, DRAFTING TOOL TO PLOT PLANE STRUCTURES       R373
BAPL 6600 F4     RS P XT O$ REDUX, REACTOR FLUCTUATION EXPERIMENT ANALYSIS     R425
BAPL 6600 F4     RS P XT P$ CAFT1, LEAST SQUARES FIT FISSILE NUCLIDE DATA      R327
BAPL 6600 F4     RS P XT P$ M0661,M0657,M0626, POLYNOMIAL CURVE FITTING        R411
BC    625 F4     RS    T C$ EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PROGRAM       156
BC    625 F4     RS    T C$ FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE       28
BC    625 F4     RS P  T C$ DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM        209
BC    625 F4     RS P XT B$ GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV         185
BCD   R   L T Z$ ETOG1 DATA LIBRARIES, MUFT4 OR 5 + GAM1 + GAM2 WNES 6600       447
BCL  6400 F4     RS P  T G$ ECCSA1, LOSS-OF-COOLANT + EMERGENCY COOLING         330
BCL  6400 F4     RS P  T G$ NURLOC-1.0, LOSS-OF-COOLANT THERMAL ANALYSIS        328
BEAMS    ANL  3600 F36    SBP    I$ BOW2, DEFLECTION CALCULATION PARALLEL       365
BEAMS  KAPL 6600 F4     RS P    I$ STEM, MATRIX GENERATION FOR A SYSTEM OF     R337
BE21, FEW-GP DISCRETE ORDINATES SLAB GEOMETRY  BAPL 6600 F4     RS P XT C $ R398
BHSC  360 F4     RS P     C$ PERT, 1-D PERTURBATION FOR AIM AND FOG CODES        30
BHSC  360 F4     RS P  T C$ FAIMOS, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE      120
BHSC  360 F4     RS PL T B$ TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS        50
BIN   R   L T Z$ ZPR-III ASSEMBLY 48 GAFGAR ENDF/B DATA TAPES   GGA  1108       356
BISYN, 2-D MULTI-GP DIFFUSION SYNTHESIS CALC   NED   635 F4     RS PL T C $     287
BLAST, REACTOR KINETICS TEMPERATURE DIST STUDY AEB   360 F4     RS P    E $     363
BLOOST5, POINT-KINETICS WITH 2-D HEAT TRANSFER GGA  7044 F4     RS    T H $     205
BLOOST6, COMBINED KINETICS 2-D HEAT TRANSFER   GGA  1108 F4     RS P  T E $     303
BLOWDOWN  PPCO  360 F4     RS P X  G$ BURST1, HYDRODYNAMIC ANALYSIS DURING       435
BLOWDOWN - EXCURSION ANALYSIS  INC  7044 F4     RS P  T G$ RELAP2, REACTOR       369
BLOWDOWN ANALYSIS LOFT    KE   7094 F+MAP RSBP  T G$ SATURATED BLOWDOWN2,        200
BLOWDOWN ANALYSIS LOFT    UGA   360 F4     RS P    G$ WATER-HAMMER, LIQUID       278
BLOWDOWN PRESSURE TEMPERATURE HISTORY    KE   7094 F2     RSBP  T H$ PTH1,      155
BLOWDOWN2, BLOWDOWN ANALYSIS LOFT    KE   7094 F+MAP RSBP  T G$ SATURATED       200
BL47, DRAFTING TOOL TO PLOT PLANE STRUCTURES   BAPL 6600 F4     RS P XT L $ R373
BNL  6600 F4     RS P     A$ AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC       376
BNL  6600 F4     RS P     A$ SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC         377
BNL  6600 F4     RS P  T F$ NOAH, 1-D ONE-GP SPACE-TIME DIFFUSION FEEDBACK       405
BNL  7090 F+FAP RS P  T M$ DFSR, DATA FILE SERVICE ROUTINES ENDF TAPES          236
BNL  7094 F4     RS P     A$ AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC       376
BNL  7094 F4     RS P     A$ SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC         377
BNL-DP  360 F4    RS PLXT B$ HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM        277
BNL-DP 7090 F2    RS PLXT B$ HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM        277
BNW  1107 F4     RS P  T C$ HFN, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE         241
BNW  1107 F4     RS PL T B$ GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV         185
BNW  1108 F4     RS P     C$ PERT4, 2-D PERTURBATION XY RZ RTHETA GEOMETRY      304
BNW  1108 F4     RS P  T B$ FCC4, FUNDAMENTAL MODE FAST REACTOR X-SEC CALC      306
BNW  1108 F4     RS P  T C$ 2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION         325
BNW  1108 F4     RS P XT B$ 1DX, 1-D DIFFUSION FAST X-SECTION GENERATION        374
BNW  1108 F5     RSBPL T C$ ISOGEN, RADIONUCLIDE GENERATION AND DECAY           367
BNW  7090 F+FAP RS PL T B$ GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV          185
BNW  7090 FLOCO RSBP  T C$ GE-HAPO-S13, 1-D MULTI-GP DOUBLE SN APPROX            75
BNW  7090 F2     RSBPL T J$ MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY         143
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BNWL 1108 F4    RS P  T B$ ETOX, MULTIGROUP CONSTANTS FROM ENDF/B FOR 1DX   388
BNWL 1108 F4    RS P  T H$ COBRA2, ROD BUNDLE THERMALHYDRAULIC ANALYSIS     432
BODIES  GGA  1108 F4    RS P    I$ SAFE-PLANE, PLANE STRESS ANALYSIS, 2-D   252
BOILING FLOW STEAM     GGA  1108 F4    RS P  T H$ DYNAM, DYNAMIC ANALYSIS   440
BOUNCE, FLUX DIST IN MULTI-PIN FUEL ELEMENT     AGC  7090 F2    RS P    B $  237
BOW2, DEFLECTION CALCULATION PARALLEL BEAMS     ANL  3600 F36    SBP    I $  365
BREEDERS     APO   635 F4    RS P    D$ RAPFU, FUEL CYCLE PARAMETERS FAST   372
BUCKLING CYL FUEL ELEMENTS   AEB   360 F4     S P    B$ WELWING, MATERIAL   362
BUGTRI, 2-D MULTIGP DIFFUSION+BURNUP TRI-MESH  GGA  1108 F4    RS P  T D $  439
BUG2, 2-D MULTIGROUP DIFFUSION & BURNUP XY, RZ GGA  1108 F4    RS P  T D $  438
BUILDUP FACTOR CALC    BAPL 6600 F+ASC RS P XT J$ ASPIS, GAMMA RAY SOURCE  R429
BUNDLE THERMALHYDRAULIC ANALYSIS   BNWL 1108 F4    RS P  T H$ COBRA2, ROD   432
BUNDLE THERMALHYDRAULIC ANALYSIS AJ-ANL  360 F4    RS P  T H$ COBRA2, ROD   432
BUNDLES  LASL 7094 F2    RSBP    H$ AXFLU, HEAT TRANSFER MOLTEN FUEL TUBE  182
BUNDLES  LASL 7094 F2    RSBP    H$ AXTHRM, HEAT TRANSFER SOLID FUEL TUBE  183
BURNUP ANALYSIS GEV   635 F4    RS P  T O$ BURNUP, HEAVY ELEMENT ISOTOPIC   311
BURNUP CYL LATTICE   WAPD 7094 F+MAP RS PLXT B$ LASER, SPECTRUM CALC WITH  249
BURNUP POWER DISTRIBUTION SEARCH    GGA  1108 F4    RS P  T C$ GASP7, 1-D   319
BURNUP RZ GEOMETRY   GGA  7090 F+FAP RSBP  T D$ DOB, 2-D FEW-GP DIFFUSION   99
BURNUP XY, RZ GGA  1108 F4    RS P  T D$ BUG2, 2-D MULTIGROUP DIFFUSION &  438
BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS GEV   635 F4    RS P  T O $  311
BURP1, DETECTOR EFFICIENCY POINT SOURCE         UM   7090 MAD   RSB    O $  164
BURP2, DETECTOR EFFICIENCY DISK SOURCE          UM   7090 MAD   RSB    O $  165
BURP3, DETECTOR EFFICIENCY POINT SOURCE         UM   7090 MAD   RSB    O $  166
BURP4, GAMMA-RAY PHOTOFRACTION SOLID CRYSTAL    UM   7090 MAD   RSB    O $  169
BURP5, GAMMA-RAY PHOTOFRACTION WELL CRYSTAL     UM   7090 MAD   RSB    O $  170
BURST1, HYDRODYNAMIC ANALYSIS DURING BLOWDOWN  PPCO  360 F4    RS P X  G $  435
BW   2000 F4    RS      A$ STRIP, RESOLVED RESONANCE INTEGRAL CALCULATION   305
CAESAR4,LIBLST, 1-D MULTI-GP DIFFUSION + LIB    AI   360 F4    RSBPLXT C $  270
CASCADE,CLUSTER, RADIATION DAMAGE IN METALS     GEC  635 F+FAP RSBP  T Q $  419
CAVITY GEOM GGA  1108 F4    RS P  T J$ MUSCAT, VIEW FACTOR SHIELDING CODE   259
CDC  1604 F63   RS      C$ PERT, 1-D PERTURBATION FOR AIM AND FOG CODES      30
CDC  1604 F63   RS  L T D$ SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION        58
CDC  1604 F63   RS P    A$ HAFEVER, HAUSER-FESHBACH INELASTIC SCATTERING     14
CDC  1604 F63   RS P    B$ SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY     52
CDC  1604 F63   RS P    B$ S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION       53
CDC  1604 F63   RS P    E$ AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK    121
CDC  1604 F63   RS P    G$ CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD           47
CDC  1604 F63   RS P    J$ GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY       45
CDC  1604 F63   RS P    J$ GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM     46
CDC  1604 F63   RS P  T C$ EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER SLAB     87
CDC  1604 F63   RS P  T C$ FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE    28
CDC  1604 F63   RS P  T C$ WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY     32
CDC  1604 F63   RS P  T C$ 20GRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER    40
CDC  1604 F63   RS PL T A$ ARES2, RESONANCE INTEGRAL X-SECTION CALC          89
CDC  1604 F63   RS PL T B$ FORM, FAST NEUTRON SPECTRUM X-SECTION CALC        51
CDC  1604 F63   RS PL T B$ TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS     50
CDC  1604 F63   RS PL T C$ AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE      29
CDC  1604 F63   RS PLXT B$ GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC        33
CDC  1604 F63   RS PLXT C$ FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE     120
CDC  3600 F63   RSBP    F$ AX1, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE     102
CDC  3600 F63   RSBP  T C$ FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION     167
CE    360 F4    RS P XT F$ CEXE,INCEXE, 1-GP 3-D XYZ XENON OSCILLATION      415
CELL   ANL  3600 F36   RSBP XT B$ RABBLE,WLIB,FLAT, RESONANCE ABSORPTION,   281
CELL  ANL  3600 F36   RSBP XT B$ RIFF-RAFF, RESONANCE INTEGRAL CALC 2-REG   213
CELL  BAPL 6600 F4    RS P XT B$ RESC2,RESCO,DBF1, RESONANCE INTEGRAL HEX  R285
CELL CALCULATION   AI    360 F4    RS P XT B$ HWOCR-SAFE, 2-D MONTE CARLO   307
CELL CODE AND TEMPEST COMBINATION  AI   7090 F2    RS P    B$ BAM, S4 CYL   108
CELL CODE, 1-D 1-GP S4 APPROXIMATION    AI   7090 F2    RS      B$ S4 CYL    53
CELL CODE, 1-D 1-GP S4 APPROXIMATION    CDC  1604 F63   RS P    B$ S4 CYL    53
CENTRIFUGAL PUMP IMPELLER DESIGN STUDY    PW   1604 F63   RS P  T I$ PIP,    187
CEXE,INCEXE, 1-GP 3-D XYZ XENON OSCILLATION     CE   360 F4    RS P XT F $  415
CHAC, LEGENDRE COEF CALC FOR ANGULAR DIST DATA AI   7094 F+MAP RS P    A $  215
CHAIN ANALYSIS        ORNL 1604 F63   RS P    D$ ISOCRUNCH, REACTION DECAY  180
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CHAIN ANALYSIS ORNL 7090 F2 RS P D$ ISOCRUNCH, REACTION DECAY 180
CHAIN EQUATIONS ORNL 7090 F2 RSBP D$ NUCY, SOLUTION OF NUCLIDE 134
CHAINS GEC 635 F4 RSBP D$ CHAINS, ANALYSIS OF RADIOACTIVE DECAY 418
CHAINS, ANALYSIS OF RADIOACTIVE DECAY CHAINS GEC 635 F4 RSBP D $ 418
CHECKER,CRECT,CAMMET,PLOTFB,SLAV3, ENDF/B PROC NCSC 6600 F4 RS T M $ 384
CHEMLOC2, CORE HEATING CLADDING-STEAM REACTION ANL 3600 F36 RSBP G $ 366
CHORD DIST FUNCT AEG 7090 F2 RS B$ DANCOFF JR, MODERATOR SPACE 150
CINCAS, NUCLEAR FUEL CYCLE COST AND ECONOMICS COMM 360 F4 RS P T D $ 354
CINDER,MO102, POINT DEPLETION FISSION PRODUCT BAPL 6600 F4 RS P D $ 313
CITATION, 1,2,3-D DIFFUSION DEPLETION MULTIGP ORNL 360 F4 RS P XT K $ 387
CLADDING-STEAM REACTION ANL 3600 F36 RSBP G$ CHEMLOC2, CORE HEATING 366
CLIP, FORM OR THREDES LIBRARY UTILITY ROUTINE AI 360 F4 RSB L T M $ 271
CLOUD AI 7090 F2 RS P G$ CLOUD, GAMMA-RAY DOSE RATE FROM A 47
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CLOUD DP 360 F4 RS P G$ CLOUD, GAMMA-RAY DOSE RATE FROM A 47
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DIFFUSION      UMCC  360 F4     RS P  T C$ VARI-QUIR, TIME-DEP 2-D MULTI-GP   212
DIFFUSION      WANL 6600 F4     RS P  T C$ VARI-QUIR, TIME-DEP 2-D MULTI-GP   212
DIFFUSION + LIB   AI    360 F4     RSBPLXT C$ CAESAR4,LIBLST, 1-D MULTI-GP   270
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DIFFUSION AND CEPLETICN     ANL  3600 F4     RSBP  T D$ 2DB, 2-D MULTIGROUP   325
DIFFUSION AND CEPLETICN     BNW  1108 F4     RS P  T D$ 2DB, 2-D MULTIGROUP   325
DIFFUSION AND CEPLETICN     IBM   36C F+BAL RS P  T D$ PDQ5, 2-D FEW-GROUP  R336
DIFFUSION AND CEPLETICN     LASL 6600 F4     RS P  T D$ 2DB, 2-D MULTIGROUP   325
DIFFUSICN AND CEPLETICN GGA  1108 F4     RS P  T D$ FEVER7, 1-D MULTIGROUP   318
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DIFFUSION CEPLETION      AI    360 F4     RSBP XT D$ SIZZLE, 1-D MULTIGROUP   58
DIFFUSICN CEPLETION      AI   7090 F2     RS  L T D$ SIZZLE, 1-D MULTIGROUP   58
DIFFUSICN CEPLETION      CDC  1604 F63    RS  L T D$ SIZZLE, 1-D MULTIGROUP   58
DIFFUSION CEPLETION     BAPL 6600 F+ASC RS P XT D$ PDQ7, 1,2 OR 3-D FEW-GP R275
DIFFUSION CEPLETION     IBM   360 F4     RS P  T C$ PDQ7, 1,2 CR 3-D FEW-GP R275
DIFFUSION DEPLETION   GGA  7044 F4     RS P  T D$ RELOAD-FEVER, 1-D FEW-GP   221
DIFFUSION DEPLETION CCDE ORNL 7090 F+FAP RS P  T D$ ASSAULT, 2-D MULTI-GP   240
DIFFUSION CEPLETION GGA  1108 F4     RS P  T D$ GAUGE, 2-D FEW-GP HEX GEOM   339
DIFFUSION CEPLETION MULTIGP  ORNL  360 F4     RS P XT K$ CITATICN, 1,2,3-D   387
DIFFUSICN CEPLETION PROGRAM  GGA  7090 F2     RSB   T C$ FEVER, 1-D FEW-GP   117
DIFFUSION CESIGN SYSTEM    AI    360 F4     RSBP  T K$ THREDES, 1-D FEW-GP   273
DIFFUSION FAST X-SECTION GENERATION    BNW  1108 F4     RS P XT B$ 1DX, 1-D   374
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DIFFUSION FEEDBACK  BAPL 6600 F4    RS P XT F$ TWIGL, 2-D 2-GP SPACE-TIME  R338
DIFFUSION FEEDBACK BNL  6600 F4    RS P  T F$ NOAH, 1-D CNE-GP SPACE-TIME   405
DIFFUSION GGA  1108 F4    RS P  T F$ GAKIN, 1-D MULTIGROUP TIME-DEPENDENT  310
DIFFUSION POWER DIST SEARCH  GGA  7044 F4    RS P  T C$ GASP2, 1-D FEW-GP   220
DIFFUSION PROGRAM   AAEC  360 F+BAL RS P  T C$ CRAM, 1-D AND 2-D MULTI-GP   103
DIFFUSION PROGRAM   ORNL 7090 F+FAP RS P  T C$ EXTERMINATOR, 2-D MULTI-GP   156
DIFFUSION PROGRAM   UK-R 7090 F+FAP RSBPL T C$ CRAM, 1-D AND 2-D MULTI-GP   103
DIFFUSION PROGRAM  BC    625 F4    RS    T C$ EXTERMINATOR2, 2-D MULTI-GP  156
DIFFUSION PROGRAM  ORNL  360 F4    RS P  T C$ EXTERMINATOR2, 2-D MULTI-GP   156
DIFFUSION SLAB CYL     ORNL 7090 F2    RS P  T C$ EQUIPOISE3, 2-D 2-GROUP    39
DIFFUSION SLAB CYL SPHERE    BNW  1107 F4    RS P  T C$ HFN, 1-D MULTI-GP   241
DIFFUSION SLAB CYL SPHERE   AI   7090 F+FAP RS  L T C$ FAIM, 1-D MULTI-GP   120
DIFFUSION SLAB CYL SPHERE   AI   7090 F+FAP RS PL T C$ AIM6, 1-D MULTI-GP    29
DIFFUSION SLAB CYL SPHERE   CDC  1604 F63   RS PL T C$ AIM6, 1-D MULTI-GP    29
DIFFUSION SLAB CYL SPHERE   CDC  1604 F63   RS PLXT C$ FAIM, 1-D MULTI-GP   120
DIFFUSION SLAB CYL SPHERE  AI   7090 F+FAP RS  L T C$ ULCER, 1-D MULTI-GP   118
DIFFUSION SLAB CYL SPHERE  ANL  3600 F36   RS P XT C$ MACH1, 1-D MULTI-GP   262
DIFFUSION SLAB CYL SPHERE  PURD 6500 F4    RS PLXT C$ MACH1, 1-D MULTI-GP   262
DIFFUSION SLAB CYL SPHERE BHSC  360 F4    RS P  T C$ FAIMOS, 1-D MULTI-GP   120
DIFFUSION SLAB CYLINDER CDC  1604 F63   RS P  T C$ 20GRAND, 2-D FEW-GROUP    40
DIFFUSION SLAB CYLINDER ORNL 7090 F2    RS P  T C$ 20GRAND, 2-D FEW-GROUP    40
DIFFUSION SLAB CYLINDER SPHERE AI   7090 F2    RS P  T C$ FOG, 1-D FEW-GP    28
DIFFUSION SLAB CYLINDER SPHERE BC    625 F4    RS    T C$ FOG, 1-D FEW-GP    28
DIFFUSION SLAB CYLINDER SPHERE CDC  1604 F63   RS P  T C$ FOG, 1-D FEW-GP    28
DIFFUSION SLAB, SPH, CYL GGA  1108 F4    RS P  T C$ GAZE2, 1-D MULTIGROUP   430
DIFFUSION SYNTHESIS CALC   NED   635 F4    RS PL T C$ BISYN, 2-D MULTI-GP   287
DIFFUSION TEMP COEF CALC     GGA  7044 F4    RS P  T C$ TEMCO, 1-D FEW-GP   225
DIFFUSION TRIANGULAR MESH  GGA  1108 F+BAL RS P  T C$ GAMTRI, 2-D MULTIGP   401
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DIFFUSION XY RZ GEOMETRY GGA  7044 F+MAP RSBP  T C$ GAMBLE4, 2-D MULTI-GP   222
DIFFUSION XY RZ RTH   WANL 7094 F4    RS P XT C$ VARI-QUIR3, 2-D MULTI-GP   264
DIFFUSION XYZ GEOMETRY  CDC  1604 F63   RS P  T C$ WHIRLAWAY, 3-D 2-GROUP    32
DIFFUSION XYZ GEOMETRY  ORNL 7090 F2    RS P  T C$ WHIRLAWAY, 3-D 2-GROUP    32
DIFFUSION 3-GEOM    BAPL 6600 F4    RS P XT F$ WIGL2, 1-D 2-GP SPACE-TIME  R274
DIFFUSION 3-GEOM    GGA  1108 F4    RS P  T F$ WIGL2, 1-D 2-GP SPACE-TIME  R274
DIFFUSION+BURNUP TRI-MESH  GGA  1108 F4    RS P  T D$ BUGTRI, 2-D MULTIGP   439
DISCRETE ORDINATE CALC      AI   7090 F2    RS PLXT C$ DTF2, 1-D MULTI-GP   151
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ETOG1, ENDF/B TO MUFT, GAM, ANISN X-SEC FORMAT WNES 6600 F4     RS PL T B $  437
ETOM1, ENDF/B FORMAT TO MUFT FORMAT X-SECTIONS WNES 6600 F4     RS P  T B $  436
ETOX, MULTIGROUP CONSTANTS FROM ENDF/B FOR 1DX BNWL 1108 F4     RS P  T B $  388
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EXCURSION CALCULATIONS      NED   635 F4     RS P XT F$ FORE2, FAST REACTOR   174
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EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PROGRAM  BC    625 F4     RS     T C $  156
EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PROGRAM  ORNL   360 F4     RS P  T C $  156
EXTERNAL+INTERNAL DOSE AI   7094 F+FAP RS PL T G$ CURIE,DOSE,THUNDERHEAD,    196
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F+ASC RS P  T L$ DATATRAN 2-D GEOMETRY INPUT, PREPARATION, EDIT KAPL 6600  R406
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F+ASC RS P  T P$ SNEQ, NONLINEAR ALGEBRAIC EQN SOLN CURVE PLOT  KAPL 6600  R364
F+ASC RS P XT D$ PDQ7, 1,2 OR 3-D FEW-GP DIFFUSION DEPLETION    BAPL 6600  R275
F+ASC RS P XT J$ ASPIS, GAMMA RAY SOURCE BUILDUP FACTOR CALC    BAPL 6600  R429
F+BAL RS  L T M$ SCORE2, SCISRS ENDF/B GRAPHIC X-SEC EVALUATION AI    360   375
F+BAL RS P  T B$ RAFFLE, 1ST FLIGHT COLLISION PROBABILITIES MC  ORNL  360   392
F+BAL RS P  T C$ CRAM, 1-D AND 2-D MULTI-GP DIFFUSION PROGRAM   AAEC  360   103
F+BAL RS P  T C$ GAMBLE5, 2-D MULTI-GP DIFFUSION XY RZ GEOMETRY GGA  1108   222
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F+BAL RS P  T E$ AIRCS2A, SIMULATION OF REACTOR DYNAMICS        AI    360   326
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F+CDP RS P XT B$ FMC-N,FMC-G, MC NEUTRON, GAMMA-RAY HISTORIES   PW   1604   195
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F+CCM RS P XT I$ GAPL3, INELASTIC LARGE DEFLECTION STRESS STUDY BAPL 6600  R397
F+CCM RSB   T M$ DATATRAN, MODULAR PROGRAMMING AND DATA SYSTEM  KAPL 6600  R386
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F+FAP RS  L T B$ QUICKIE, INFINITE MEDIUM SPECTRUM X-SECTIONS   AI   7090   119
F+FAP RS  L T C$ FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE   AI   7090   120
F+FAP RS  L T C$ ULCER, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE  AI   7090   118
F+FAP RS  L T M$ MOMUS, X-SECTION LIBRARY UTILITY PROGRAM       AI   7094   159
F+FAP RS P  T B$ EPITHERMOS, SPECTRUM AND X-SECTION CALCULATION GEV  7094   201
F+FAP RS P  T B$ RAFFLE, 1ST FLIGHT COLLISION PROBABILITIES MC  ORNL 7090   392
F+FAP RS P  T B$ SOPHIST1/2/5, MULTI-GP TRANSFER COEFFICIENTS   LRL  7090   160
F+FAP RS P  T C$ EXTERMINATOR, 2-D MULTI-GP DIFFUSION PROGRAM   ORNL 7090   156
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F+FAP RS P  T D$ ASSAULT, 2-D MULTI-GP DIFFUSION DEPLETION CODE ORNL 7090   240
F+FAP RS P  T G$ AISITE2, PARAMETRIC SITE REQUIREMENT STUDY     AI   7094   172
F+FAP RS P  T M$ DFSR, DATA FILE SERVICE ROUTINES ENDF TAPES    BNL  7090   236
F+FAP RS PL T B$ AILMOE, X-SECTION CALC ELASTIC SCAT RESONANCES AI   7094   147
F+FAP RS PL T B$ GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV    BNW  7090   185
F+FAP RS PL T B$ GRAVE, GROUP-AVERAGING X-SECTIONS PARAMETERS   AI   7094   162
F+FAP RS PL T B$ TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS  AI   7090    50
F+FAP RS PL T B$ TYCHE3, MONTE CARLO SLOWING-DOWN DENSITY CALC  AI   7094   149
F+FAP RS PL T C$ AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE   AI   7090    29
F+FAP RS PL T G$ CURIE,DOSE,THUNDERHEAD, EXTERNAL+INTERNAL DOSE AI   7094   196
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F+FAP RSB   T A$ ZUT, RESOLVED REGION RESONANCE INTEGRAL CALC   GGA 7090    41
F+FAP RSBP  T D$ DDB, 2-D FEW-GP DIFFUSION BURNUP RZ GEOMETRY   GGA 7090    99
F+FAP RSBP  T Q$ CASCADE,CLUSTER, RADIATION DAMAGE IN METALS    GEC  635   419
F+FAP RSBPL T C$ CRAM, 1-D AND 2-D MULTI-GP DIFFUSION PROGRAM   UK-R 7090  103
F+GMP RSBP  T H$ THTE, 3-D TRANSIENT HEAT TRANSFER PROGRAM      GEC  635   346
F+MAP RS P    A$ CHAD, LEGENDRE COEF CALC FOR ANGULAR DIST DATA AI   7094   215
F+MAP RS P  T B$ GARCL, RESONANCE OVERLAP AND LATTICE EFFECTS   GGA 7044   219
F+MAP RS P  T C$ TDSN, 2-D MULTIGROUP DISCRETE ORDINATE PROGRAM LER 7090   312
F+MAP RS P  T D$ OPUS, POWER PLANT PERFORMANCE AND PRICE STUDY  GGA 7044   226
F+MAP RS P  T E$ AIROS, SPACE-INDEPENDENT KINETICS W/FEEDBACK   AI   7094   163
F+MAP RS P  T G$ CONTEMPT, LOSS-OF-COOLANT ACCIDENT ANALYSIS    PPCO 7040   297
F+MAP RS P  T G$ RSAC, RADIOLOGICAL SAFETY ANALYSIS PROGRAM     PPCO 7040   265
F+MAP RS P  T H$ TOODEE, 2-D TIME-DEPENDENT HEAT CONDUCTION     PPCO 7044   349
F+MAP RS PLXT A$ TRIX1, RESONANCE INTEGRAL X-SECTION CALC       AI   7094   208
F+MAP RS PLXT B$ LASER, SPECTRUM CALC WITH BURNUP CYL LATTICE   WAPD 7094   249
F+MAP RSBP    M$ UNPACK, RETRIEVAL FROM SCISRS X-SECTION TAPE   GGA 7044   206
F+MAP RSBP  T C$ GAMBLE4, 2-D MULTI-GP DIFFUSION XY RZ GEOMETRY GGA 7044   222
F+MAP RSBP  T G$ SATURATED BLOWDOWN2, BLOWDOWN ANALYSIS LOFT    KE   7094   200
FACTOR CALC     BAPL 6600 F+ASC RS P XT J$ ASPIS, GAMMA RAY SOURCE BUILDUP R429
FACTOR SHIELDING CODE CAVITY GEOM GGA  1108 F4    RS P  T J$ MUSCAT, VIEW  259
FACTORS     ORNL  360 F4    RS P  T A$ ATHENA4, INELASTIC SCATTERING FORM  417
FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE   AI   7090 F+FAP RS  L T C $ 120
FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE   CDC  1604 F63   RS PLXT C $ 120
FAIMOS, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE BHSC  360 F4    RS P  T C $ 120
FAP   RS P  T H$ RATH, 2- OR 3-D HEAT CONDUCTION LUMPED MASS    LASL 7094  242
FAP   RSBP XT Q$ SPECTRA, MC CALCULATION IRRADIATED MATERIAL    GEC  7090  422
FARED, 1-D FAST REACTOR DESIGN & SURVEY STUDY  B&W  6600 F+COM RS P XT D $ 427
FARSE1A, DOSE RATE FROM SNAP SHIELD LEAKAGE    AI   7090 F2    RS      J $  91
FASDOP, X-SECTIONS FROM B-W RESONANCE DATA     GGA  7044 F4    RS P    A $ 216
FAST BREEDERS     APC   635 F4    RS P    D$ RAPFU, FUEL CYCLE PARAMETERS  372
FAST NEUTRON SPECTRUM X-SECTION CALC      AI   7090 F2    RS PL T B$ FORM,  51
FAST NEUTRON SPECTRUM X-SECTION CALC      ANL  3600 F63   RS  LXT B$ GAM1,  33
FAST NEUTRON SPECTRUM X-SECTION CALC      CDC  1604 F63   RS PL T B$ FORM,  51
FAST NEUTRON SPECTRUM X-SECTION CALC      CDC  1604 F63   RS PLXT B$ GAM1,  33
FAST NEUTRON SPECTRUM X-SECTION CALC      GGA  7090 F2    RS PLXT B$ GAM1,  33
FAST REACTOR DESIGN & SURVEY STUDY  B&W  6600 F+COM RS P XT D$ FARED, 1-D 427
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FAST REACTOR EXCURSION CALCULATIONS       NED  635 F4     RS P XT F$ FORE2, 174
FAST REACTOR POWER AND FLOW TRANSIENTS    ANL  360 F4     RS P  T F$ SAS1A, 400
FAST REACTOR POWER AND FLOW TRANSIENTS    ANL  6600 F4    RS P  T F$ SAS1A, 400
FAST REACTOR X-SEC CALC BNW  1108 F4    RS P  T B$ FCC4, FUNDAMENTAL MODE 306
FAST SPECTRUM MULTI-GP CONSTANT CALC  AGC  7090 F2    RSBPLXT B$ AGN-GAM, 204
FAST THERMAL SPECTRA GGA  1108 F4    RS PLXT B$ GGC4, MULTI-GP X-SECTIONS 298
FAST THERMAL SPECTRA GGA  6600 F4    RS  L T B$ GGC4, MULTI-GP X-SECTIONS 298
FAST X-SECTION GENERATION    BNW  1108 F4    RS P XT B$ 1DX, 1-D DIFFUSION 374
FATIGUE     KAPL 6600 F4    RS      I$ SORSDB, PRESSURE VESSEL STRESS AND R391
FCC4, FUNDAMENTAL MODE FAST REACTOR X-SEC CALC BNW  1108 F4    RS P  T B $ 306
FEEDBACK  BAPL 6600 F4    RS P XT F$ NOWIG, 1-D 2-GP KINETICS TEMPERATURE R371
FEEDBACK  BAPL 6600 F4    RS P XT F$ TWIGL, 2-D 2-GP SPACE-TIME DIFFUSION R338
FEEDBACK BNL  6600 F4    RS P  T F$ NOAH, 1-D ONE-GP SPACE-TIME DIFFUSION 405
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FEVER7, 1-D MULTIGRCLP DIFFUSICN AND DEPLETION GGA  1108 F4     RS P  T D $  318
FEW-GP DIFFUSICN BURNUP RZ GECMETRY   GGA  7090 F+FAP RSBP  T D$ DDB, 2-D    99
FEW-GP DIFFUSICN CALC HEX-Z MESH      GGA  1108 F4     RS P  T C$ GATT, 3-D  380
FEW-GP DIFFUSION DEPLETICN     BAPL 6600 F+ASC RS P XT D$ PDQ7, 1,2 OR 3-D  R275
FEW-GP DIFFUSION DEPLETICN     IBM  360 F4     RS P  T D$ PDQ7, 1,2 OR 3-D  R275
FEW-GP DIFFUSION DEPLETICN     GGA  7044 F4    RS P  T D$ RELOAD-FEVER, 1-D  221
FEW-GP DIFFUSION DEPLETICN PROGRAM  GGA  7090 F2    RSB   T D$ FEVER, 1-D  117
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FEW-GP DIFFUSICN POWER DIST SEARCH  GGA  7C44 F4    RS P  T C$ GASP2, 1-D  220
FEW-GP DIFFUSICN SLAB CYLINDER SPHERE AI   7090 F2    RS P  T C$ FOG, 1-D   28
FEW-GP DIFFUSICN SLAB CYLINDER SPHERE BC    625 F4    RS    T C$ FOG, 1-D   28
FEW-GP DIFFUSICN SLAB CYLINDER SPHERE CDC  1604 F63   RS P  T C$ FOG, 1-D   28
FEW-GP DIFFUSION TEMP CCEF CALC     GGA  7044 F4    RS P  T C$ TEMCO, 1-D  225
FEW-GP DISCRETE GRDINATES SLAB GEOMETRY  BAPL 6600 F4    RS P XT C$ BE21,  R398
FEW-GP HEX GECM DIFFUSICN DEPLETICN GGA  1108 F4    RS P  T D$ GAUGE, 2-D  339
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FEW-GP SN DOUBLE SN APPROX SLAB GECM PPCO 7090 F2    RS P    C$ MIST, 1-D   59
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FEW-GROUP DIFFUSION AND DEPLETICN   IBM  360 F+BAL RS P  T D$ PDQ5, 2-D  R336
FEW-GROUP DIFFUSION SLAB CYLINDER CDC  1604 F63   RS P  T C$ 20GRAND, 2-D   40
FEW-GROUP DIFFUSION SLAB CYLINDER CRNL 7090 F2    RS P  T C$ 20GRAND, 2-D   40
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FINITE-ELEMENT STUDY 2,3-D STRUCTURES   KAPL 6600 F4    RS P X  I$ FINEL,  R404
FINS SNAP GEOM   AI   7C90 F2    RS      J$ SCARF2, SCATTER FROM RADIATOR  110
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FISSILE NUCLIDE DATA  BAPL 6600 F4    RS P XT P$ DAFT1, LEAST SQUARES FIT  R327
FISSILE NUCLIDE X-SECTICN EVALUATION   CRNL 7090 F4    RSBP    A$ MUFFLE,  323
FISSICN PRCDUCT  BAPL 66CO F4    RS P    D$ CINDER,M0102, POINT DEPLETION  313
FISSICN PRCDUCT ACTIVITY DIST STUDY GGA  7C44 F4    RS P    D$ RAD2, HTGR  231
FISSICN PRCDUCT RELEASE GGA  1108 F4    RS P    D$ FREVAP6, HTGR METALLIC  301
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FLAC, STEACY-STATE FLCW, PRESSURE DISTRIBUTION GGA  1108 F5    RS P    H $  395
FLANGE1, SCATTERING LAW X-SECTICN CALCULATION  GGA  1108 F5    RS P    A $  247
FLANGE2, ENDF/B THERMAL SCATTERING CATA PROC   DP    360 F4    RS P  T A $  368
FLANGE2, ENDF/B THERMAL SCATTERING CATA PROC   GGA  1108 F4    RS    T A $  368
FLARE, 3-D REACTIVITY AND POWER DISTRIBUTICN   CDC  3600 F63   RSBP  T C $  167
FLARE, 3-D REACTIVITY ANC POWER DISTRIBUTICN   NED   635 F4    RS P  T C $  167
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FLASH4, FULLY-IMPLICIT TRANSIENT SIMULATICN    BAPL 6600 F4    RS P XT G $ R448
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FLOW STEAM     GGA  11C8 F4    RS P  T H$ DYNAM, DYNAMIC ANALYSIS BOILING  440
FLOW STUDY STEAM-WATER MIX KAPL 66CC F4    RS P    H$ RAPP, HIGH-VELOCITY R382
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FOG CODES BHSC 360 F4 RS P C$ PERT, 1-D PERTURBATION FOR AIM AND 30
FOG CODES CDC 1604 F63 RS C$ PERT, 1-D PERTURBATION FOR AIM AND 30
FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE AI 7090 F2 RS P T C $ 28
FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE BC 625 F4 RS T C $ 28
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F2 RS B$ DANCOFF JR, MODERATOR SPACE CHORD DIST FUNCT AEG 7090 150
F2 RS B$ S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION AI 7090 53

F2	RS		C$	FIRN, 2-D FEW-GP S4 APPROXIMATION RZ GEOMETRY	LRL	709	7
F2	RS		C$	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES	AI	7090	30
F2	RS		E$	AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK	AI	7090	121
F2	RS		E$	SNAPKIN5/5A, 1-REGION KINETICS SNAP GEOMETRY	AI	7090	122
F2	RS		I$	CROCK, SPACE POWER PLANT DESIGN OPTIMIZATION	AI	7090	112
F2	RS		I$	FUGUE, STEADY-STATE TEMPERATURE VOID FRACTION	AI	7090	48
F2	RS		I$	SHOCK, SPACE POWER PLANT DESIGN OPTIMIZATION	AI	7090	114
F2	RS		J$	FARSE1A, DOSE RATE FROM SNAP SHIELD LEAKAGE	AI	7090	91
F2	RS		J$	MORTIMER, DOSE RATE CALCULATION SNAP GEOMETRY	AI	7090	142
F2	RS		J$	SCARF2, SCATTER FROM RADIATOR FINS SNAP GEOM	AI	7090	110
F2	RS		J$	SCAR1, SCATTER FROM A RING SNAP GEOMETRY	AI	7090	111
F2	RS		N$	WED, W-DSN OUTPUT TAPE EDIT REACTION RATES	UK-W	7090	133
F2	RS	T	A$	SUMMIT, CRYSTALLINE SCATTERING KERNEL CALC	GGA	7090	56
F2	RS	L T	D$	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	AI	7090	58
F2	RS P		A$	DOPIE, RESOLVED RESONANCE X-SECTION CALC	NED	2000	177
F2	RS P		A$	EXT, X-SECTIONS FROM B-W RESONANCE PARAMETERS	WANL	7094	238
F2	RS P		A$	RAPTURE, RESONANCE INTEGRAL X-SECTION CALC	NED	2000	176
F2	RS P		B$	BAM, S4 CYL CELL CODE AND TEMPEST COMBINATION	AI	7090	108
F2	RS P		B$	BOUNCE, FLUX DIST IN MULTI-PIN FUEL ELEMENT	AGC	7090	237
F2	RS P		B$	SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY	AI	7090	52
F2	RS P		B$	SPARTA, SPATIALLY-AVERAGED DOPPLER EFFECTS	NED	2000	178
F2	RS P		C$	MIST, 1-D FEW-GP SN DOUBLE SN APPROX SLAB GEOM	PPCO	7090	59
F2	RS P		D$	AIMFIRE, URANIUM FUEL CYCLE COST ANALYSIS	AI	7090	55
F2	RS P		D$	ISOCRUNCH, REACTION DECAY CHAIN ANALYSIS	ORNL	7090	180
F2	RS P		F$	CONEC, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE	LRL	7090	129
F2	RS P		F$	WEAK EXPLOSION, COUPLED NEUTRON-HYDRODYNAMICS	APDA	7094	145
F2	RS P		G$	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	AI	7090	47
F2	RS P		H$	FLOW-MODEL, MULTI-CHANNEL 2-D 2-PHASE FLOW	AI	7094	246
F2	RS P		I$	WIREX, COMPUTER-PRODUCED WIRING LISTS UHTREX	LASL	7090	315
F2	RS P		J$	GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY	AI	7090	45
F2	RS P		J$	GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM	AI	7090	46
F2	RS P		J$	RATRAP, DOSE RATE CALCULATION SNAP GEOMETRY	AI	7090	141
F2	RS P		J$	SHOE, SHIELD WEIGHT OPTIMIZATION DOSE CALC	AI	7094	197
F2	RS P		O$	CROC90, ML-1 FLUID FLOW EXPERIMENT ANALYSIS	AGC	7090	154
F2	RS P		P$	EXPALS, LEAST SQUARES EXPONENTIAL DECAY CURVES	LRL	7094	321
F2	RS P		P$	FRANTIC, LEAST SQUARES FIT SUM OF EXPONENTIALS	MIT	7090	324
F2	RS P	T	A$	2PLLS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS	AI	7094	254
F2	RS P	T	B$	AGN-SIGMA, CALC OF MULTI-GP TRANSFER MATRICES	AGC	7090	243
F2	RS P	T	C$	EQUIPOISE3, 2-D 2-GROUP DIFFUSION SLAB CYL	ORNL	7090	39
F2	RS P	T	C$	EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER SLAB	ORNL	7090	87
F2	RS P	T	C$	FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE	AI	7090	28
F2	RS P	T	C$	TOPIC, 1-D FEW-GP SN APPROXIMATION CYLINDER	PPCO	7090	148
F2	RS P	T	C$	W-DSN, 1-D MULTI-GP SN APPROX SLAB CYL SPHERE	UK-W	7090	132
F2	RS P	T	C$	WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY	ORNL	7090	32
F2	RS P	T	C$	20GRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER	ORNL	7090	40
F2	RS P	T	G$	TRANS-FUGUE1, TRANSIENT FLOW AND HEAT TRANSFER	AI	7094	268
F2	RS P	T	H$	HEATING2, TRANSIENT STEADY-STATE HEAT TRANSFER	AI	7094	198
F2	RS P	T	L$	DPC, DATA PREPARATION FOR 2-D DESIGN PROGRAMS	LASL	7090	234
F2	RS PL	T	A$	ARES2, RESONANCE INTEGRAL X-SECTION CALC	AI	7090	89
F2	RS PL	T	B$	FORM, FAST NEUTRON SPECTRUM X-SECTION CALC	AI	7090	51
F2	RS PL	T	I$	CYCLOPS1, THERMODYNAMIC CYCLE ANALYSIS	AI	7094	244
F2	RS PL	T	M$	ECSIL, EXPERIMENTAL NEUTRON DATA LIBRARY	LRL	7094	351
F2	RS PLXT		B$	GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC	GGA	7090	33
F2	RS PLXT		B$	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM	BNL-DP	7090	277
F2	RS PLXT		C$	DTF2, 1-D MULTI-GP DISCRETE ORDINATE CALC	AI	7090	151
F2	RSB		F$	HATCHET, COUPLED NEUTRONICS-HYDRODYNAMICS CODE	AGC	7090	153
F2	RSB	T	D$	FEVER, 1-D FEW-GP DIFFUSION DEPLETION PROGRAM	GGA	7090	117
F2	RSB	T	N$	CPS, SC4020 PLOTS FROM SCISRS X-SECTION TAPES	WANL	7094	239
F2	RSBP		D$	NPRFCCP, FUEL CYCLE COSTS PERFORMANCE DATA	KE	7090	146
F2	RSBP		D$	NUCY, SOLUTION OF NUCLIDE CHAIN EQUATIONS	ORNL	7090	134
F2	RSBP		H$	AXFLU, HEAT TRANSFER MOLTEN FUEL TUBE BUNDLES	LASL	7094	182
F2	RSBP		H$	AXTHRM, HEAT TRANSFER SOLID FUEL TUBE BUNDLES	LASL	7094	183
F2	RSBP		J$	LIPRECAN1, MC NEUTRON PENETRATION CALCULATION	DAC	7090	123
F2	RSBP	T	H$	PTH1, BLOWDOWN PRESSURE TEMPERATURE HISTORY	KE	7094	155

F2	RSBP	X	C$	HERESY1, LATTICE PARAMETERS HETEROGENEOUS CALC	FMA	7090	136
F2	RSBPL	T	J$	MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY	BNW	7090	143
F2	RSBPL	XT	B$	AGN-GAM, FAST SPECTRUM MULTI-GP CONSTANT CALC	AGC	7090	204
F36	SBP		I$	BOW2, DEFLECTION CALCULATION PARALLEL BEAMS	ANL	3600	365
F36	RS P	T	B$	MC**2, ENDF MULTIGROUP X-SECTION CALCULATION	ANL	3600	355
F36	RS P	T	C$	SNARG-1D, 1-D MULTI-GP DISCRETE ORDINATE CALC	ANL	3600	288
F36	RS P	T	I$	SUPCRAN, REACTOR CORE SUPPORT STRESS ANALYSIS	ANL	3600	357
F36	RS P	XT	C$	MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	ANL	3600	262
F36	RSBP		A$	MISH-MASH, RESONANCE INTEGRAL CALC HOMOGENEOUS	ANL	3600	214
F36	RSBP		G$	CHEMLOC2, CORE HEATING CLADDING-STEAM REACTION	ANL	3600	366
F36	RSBP		G$	WHAM, LIQUID-FILLED PIPING SYSTEM ANALYSIS	KE	3600	278
F36	RSBP		I$	SWELL2, FUEL ELEMENT LIFETIME ANALYSIS	ANL	3600	353
F36	RSBP	XT	B$	RABBLE,WLIB,FLAT, RESONANCE ABSORPTION, CELL	ANL	3600	281
F36	RSBP	XT	B$	RIFF-RAFF, RESONANCE INTEGRAL CALC 2-REG CELL	ANL	3600	213
F4	S	T	D$	GAFFE, EQUILIBRIUM FUEL CYCLE CALCULATION	GGA	1108	302
F4	S P		B$	WELWING, MATERIAL BUCKLING CYL FUEL ELEMENTS	AEB	360	362
F4	RS		A$	STRIP, RESOLVED RESONANCE INTEGRAL CALCULATION	BW	2000	305
F4	RS		B$	HEXSCAT, ELASTIC SCAT X-SECTIONS HEX LATTICE	GGA	1108	291
F4	RS		E$	R102, SPACE-INDEPENDENT INVERSE KINETICS CALC	WANL	7094	168
F4	RS		I$	CORE, CORE CONFIGURATION FUEL TEMPERATURE CODE	GGA	7044	233
F4	RS		I$	SORSDB, PRESSURE VESSEL STRESS AND FATIGUE	KAPL	6600	R391
F4	RS	T	A$	FLANGE2, ENDF/B THERMAL SCATTERING DATA PROC	GGA	1108	368
F4	RS	T	C$	EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PROGRAM	BC	625	156
F4	RS	T	C$	FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE	BC	625	28
F4	RS	T	G$	GADOSE,DOSET, HTGR ACCIDENT ANALYSIS DOSE CALC	GGA	1108	261
F4	RS	T	H$	BLOCST5, POINT-KINETICS WITH 2-D HEAT TRANSFER	GGA	7044	205
F4	RS	T	H$	RATH, 2- OR 3-D HEAT CONDUCTION LUMPED MASS	LASL	7030	242
F4	RS	T	M$	CHECKER,CRECT,CAMMET,PLOTFB,SLAV3, ENDF/B PROC	NCSC	6600	384
F4	RS	T	P$	LASL LEAST SQUARES, GENERAL CURVE FITTING	LASL	7094	62
F4	RS	XT	B$	MO807, 2-D DIFFUSION ABSORPTION REMOVAL X-SECS	BAPL	6600	R280
F4	RS	XT	C$	MO648, 1-D SLAB TRANSPORT WITH SLOWING DOWN	BAPL	6600	R342
F4	RS	XT	G$	PRECON, HTGR CONTAINMENT PRESSURE POST RUPTURE	GGA	7044	228
F4	RS	XT	H$	WATER, STEAM TABLES 14.5-14,500PSIA 32-472DEGF	BAPL	6600	R267
F4	RS	XT	I$	CYGRO2, STRESS ANALYSIS CYL FUEL ELEMENT	BAPL	6600	R266
F4	RS	L T	B$	GGC4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA	GGA	6600	298
F4	RS	L T	C$	FIRE5, 1-D AGE-DIFFUSION SLAB CYLINDER SPHERE	AEB	360	9
F4	RS P		A$	AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC	BNL	6600	376
F4	RS P		A$	AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC	BNL	7094	376
F4	RS P		A$	CODILLI, LEAST SQUARES ANALYSIS RESONANCE DATA	UILL	360	347
F4	RS P		A$	COMBCO, COMBINED B-W MULTI-LEVEL CONVOLUTION	PPCO	7040	203
F4	RS P		A$	FASDOP, X-SECTIONS FROM B-W RESONANCE DATA	GGA	7044	216
F4	RS P		A$	GAKER, INELASTIC SCAT X-SECTION CALC MODERATOR	GGA	1108	289
F4	RS P		A$	GANDY, UNRESOLVED RESONANCE X-SECTION CALC	GGA	1108	341
F4	RS P		A$	GASKET, THERMAL SCATTERING LAW CALCULATION	GGA	1108	263
F4	RS P		A$	PSEUDO, STATISTICAL RESONANCE PARAMETER CALC	GGA	1108	292
F4	RS P		A$	SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC	BNL	6600	377
F4	RS P		A$	SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC	BNL	7094	377
F4	RS P		B$	AVOID, ANNULAR VOID X-SECTION CALCULATION	GEC	625	276
F4	RS P		B$	GAMMA-P, PRODUCTION X-SECTIONS FOR GAMMA-RAYS	GEC	7094	235
F4	RS P		B$	PARTI, OPTIMAL GROUP OR MESH COLLAPSING	TRW-MMU	6500	416
F4	RS P		C$	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES	BHSC	360	30
F4	RS P		C$	PERT4, 2-D PERTURBATION XY RZ RTHETA GEOMETRY	BNW	1108	304
F4	RS P		D$	CINDER,MO102, POINT DEPLETION FISSION PRODUCT	BAPL	6600	313
F4	RS P		D$	FREVAP6, HTGR METALLIC FISSION PRODUCT RELEASE	GGA	1108	301
F4	RS P		D$	ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC	ORNL	360	322
F4	RS P		D$	RAD2, HTGR FISSION PRODUCT ACTIVITY DIST STUDY	GGA	7044	231
F4	RS P		D$	RAPFU, FUEL CYCLE PARAMETERS FAST BREEDERS	APO	635	372
F4	RS P		D$	WAMPUM, FUEL CYCLE COSTS PERFORMANCE STUDY	GGA	7044	224
F4	RS P		E$	AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK	AEB	360	121
F4	RS P		E$	BLAST, REACTOR KINETICS TEMPERATURE DIST STUDY	AEB	360	363
F4	RS P		E$	GAPOTKIN, SPACE-INDEPENDENT REACTOR KINETICS	GGA	1108	317
F4	RS P		E$	GASA, STABILITY ANALYSIS REACTOR KINETICS EQNS	GGA	1108	290
F4	RS P		E$	R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS	WANL	7094	255
F4	RS P		F$	MARS, 2-D EXCURSION CALCULATION R-Z GEOMETRY	APDA	7094	293

F4	RS	P		F$	RAUMZEIT, 1-D TIME-DEPENDENT DIFFUSION CALC	KAPL	6600	R352
F4	RS	P		F$	STINT3, SINGLE-CHANNEL SPACE-TIME SYNTHESIS	KAPL	6600	R389
F4	RS	P		F$	TWIGGLE, 2-D 2-GP SPACE-TIME DIFFUSION	ANL	360	R338
F4	RS	P		G$	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	DP	360	47
F4	RS	P		G$	WATER-HAMMER, LIQUID BLOWDOWN ANALYSIS LOFT	UGA	360	278
F4	RS	P		H$	RAPP, HIGH-VELOCITY FLOW STUDY STEAM-WATER MIX	KAPL	6600	R382
F4	RS	P		I$	GAZELLE5, GAS-COOLED CORE THERMAL DESIGN STUDY	GGA	7044	232
F4	RS	P		I$	SAFE-CREEP, VISCOELASTIC ANALYSIS CONCRETE	GGA	1108	300
F4	RS	P		I$	SAFE-PLANE, PLANE STRESS ANALYSIS, 2-D BODIES	GGA	1108	252
F4	RS	P		I$	SAFE-SHELL, STRESS ANALYSIS THIN SHELLS	GGA	1108	253
F4	RS	P		I$	SAFE-2D, PLANE + AXISYMMETRIC STRESS ANALYSIS	GGA	1108	379
F4	RS	P		I$	STEM, MATRIX GENERATION FOR A SYSTEM OF BEAMS	KAPL	6600	R337
F4	RS	P		I$	TUBE, U-TUBE HEAT EXCHANGER STRESS ANALYSIS	KAPL	6600	R378
F4	RS	P		L$	GAVER, ENERGY INTERVAL X-SECTION AVERAGING	GGA	7044	218
F4	RS	P		L$	GRDWRK, GRID GENERATION FOR SAFE PROGRAMS	GGA	1108	296
F4	RS	P		N$	HEATMESH, GEOMETRICAL DATA HEAT TRANSFER STUDY	SLL	6600	434
F4	RS	P		N$	LARCA, FLUX-WEIGHTING OF DTF4 CROSS SECTIONS	LASL	6600	409
F4	RS	P		O$	ALPHA-M, RESOLUTION OF GAMMA RAY SPECTRA	ORNL	360	413
F4	RS	P		O$	CORGAM, UNFOLDING OF COMPLEX GAMMA-RAY SPECTRA	KSUN	360	390
F4	RS	P		P$	MOST, A MULTIDIMENSIONAL OPTIMIZATION SCHEME	KAPL	6600	R446
F4	RS	P		P$	ROPE, FINDING ROOTS OF A POLYNOMIAL	KAPL	6600	R444
F4	RS	P	T	A$	ATHENA4, INELASTIC SCATTERING FORM FACTORS	ORNL	360	417
F4	RS	P	T	A$	FLANGE2, ENDF/B THERMAL SCATTERING DATA PROC	DP	360	368
F4	RS	P	T	A$	TACASI, ANALYSIS OF RESONANCE MEASUREMENTS	GGA	1108	410
F4	RS	P	T	A$	TOR, THERMAL SCATTERING CRYSTALLINE MATERIALS	LASL	6600	360
F4	RS	P	T	B$	ETOM1, ENDF/B FORMAT TO MUFT FORMAT X-SECTIONS	WNES	6600	436
F4	RS	P	T	B$	ETOX, MULTIGROUP CONSTANTS FROM ENDF/B FOR 1DX	BNWL	1108	388
F4	RS	P	T	B$	FCC4, FUNDAMENTAL MODE FAST REACTOR X-SEC CALC	BNW	1108	306
F4	RS	P	T	B$	FMC-N, MONTE CARLO CALC NEUTRON HISTORIES	GGA	7044	195
F4	RS	P	T	B$	GARCL, RESONANCE OVERLAP AND LATTICE EFFECTS	LER	7094	219
F4	RS	P	T	B$	GLEN, GROUP CONSTANT CALC FROM TOR OUTPUT DATA	LASL	6600	361
F4	RS	P	T	B$	REAX, RESOLVED RESONANCE EPITHERMAL X-SECTIONS	NED	635	257
F4	RS	P	T	B$	SUPERTOG, ENDF/B FINE-GP CONSTANTS GENERATION	ORNL	360	431
F4	RS	P	T	C$	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	ANL	360	209
F4	RS	P	T	C$	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	BC	625	209
F4	RS	P	T	C$	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	LASL	6600	209
F4	RS	P	T	C$	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	LASL	7030	209
F4	RS	P	T	C$	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	LER	7094	209
F4	RS	P	T	C$	EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PROGRAM	ORNL	360	156
F4	RS	P	T	C$	FAIMOS, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	BHSC	360	120
F4	RS	P	T	C$	FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION	NED	635	167
F4	RS	P	T	C$	GASP2, 1-D FEW-GP DIFFUSION POWER DIST SEARCH	GGA	7044	220
F4	RS	P	T	C$	GASP7, 1-D BURNUP POWER DISTRIBUTION SEARCH	GGA	1108	319
F4	RS	P	T	C$	GATT, 3-D FEW-GP DIFFUSION CALC HEX-Z MESH	GGA	1108	380
F4	RS	P	T	C$	GAZE2, 1-D MULTIGROUP DIFFUSION SLAB, SPH, CYL	GGA	1108	430
F4	RS	P	T	C$	HFN, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	BNW	1107	241
F4	RS	P	T	C$	TEMCO, 1-D FEW-GP DIFFUSION TEMP COEF CALC	GGA	7044	225
F4	RS	P	T	C$	TEMCO7, TEMPERATURE COEFFICIENT CALCULATION	GGA	1108	320
F4	RS	P	T	C$	TOPIC, 1-D FEW-GP SN APPROXIMATION CYLINDER	PPCO	7040	148
F4	RS	P	T	C$	TWOTRAN, 2-D MULTI-GP TRANSPORT CODE R-Z GEOM	LASL	1108	358
F4	RS	P	T	C$	TWOTRAN, 2-D MULTI-GP TRANSPORT CODE X-Y GEOM	ANL	360	358
F4	RS	P	T	C$	TWOTRAN, 2-D MULTI-GP TRNSPT CODE XY RZ RTHETA	LASL	6600	358
F4	RS	P	T	C$	VARI-QUIR, TIME-DEP 2-D MULTI-GP DIFFUSION	UMCC	360	212
F4	RS	P	T	C$	VARI-QUIR, TIME-DEP 2-D MULTI-GP DIFFUSION	WANL	6600	212
F4	RS	P	T	D$	BUGTRI, 2-D MULTIGP DIFFUSION+BURNUP TRI-MESH	GGA	1108	439
F4	RS	P	T	D$	BUG2, 2-D MULTIGROUP DIFFUSION & BURNUP XY, RZ	GGA	1108	438
F4	RS	P	T	D$	CINCAS, NUCLEAR FUEL CYCLE COST AND ECONOMICS	COMM	360	354
F4	RS	P	T	D$	DTF-BURN, 1-D MULTI-GP DTF4 WITH DEPLETION	LASL	7030	269
F4	RS	P	T	D$	FEVER7, 1-D MULTIGROUP DIFFUSION AND DEPLETION	GGA	1108	318
F4	RS	P	T	D$	GARGOYLE, FUEL CYCLE ANALYSIS PARTIAL REFUEL	GGA	7044	260
F4	RS	P	T	D$	GAUGE, 2-D FEW-GP HEX GEOM DIFFUSION DEPLETION	GGA	1108	339
F4	RS	P	T	D$	PDQ7, 1,2 OR 3-D FEW-GP DIFFUSION DEPLETION	IBM	360	R275
F4	RS	P	T	D$	PWCOST, REACTOR FUEL CYCLE COST CALCULATION	GGA	1108	441
F4	RS	P	T	D$	RELCAD-FEVER, 1-D FEW-GP DIFFUSION DEPLETION	GGA	7044	221

F4	RS	P	T	D$	REVISED-GAD, FUEL CYCLE ANALYSIS W/REFUELING	GGA	7044	223
F4	RS	P	T	D$	STMGEN, STEAM GENERATOR DESIGN CRITERIA COSTS	GGA	7044	227
F4	RS	P	T	D$	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	BNW	1108	325
F4	RS	P	T	D$	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	LASL	6600	325
F4	RS	P	T	E$	BLCOST6, COMBINED KINETICS 2-D HEAT TRANSFER	GGA	1108	303
F4	RS	P	T	F$	FORE, FAST REACTOR EXCURSION CALCULATIONS	NED	2000	174
F4	RS	P	T	F$	GAKIN, 1-D MULTIGROUP TIME-DEPENDENT DIFFUSION	GGA	1108	310
F4	RS	P	T	F$	GAKIT, 1-D MULTIGP KINETICS WITH TEMP FEEDBACK	GGA	1108	370
F4	RS	P	T	F$	NOAH, 1-D ONE-GP SPACE-TIME DIFFUSION FEEDBACK	BNL	6600	405
F4	RS	P	T	F$	SAS1A, FAST REACTOR POWER AND FLOW TRANSIENTS	ANL	360	400
F4	RS	P	T	F$	SAS1A, FAST REACTOR POWER AND FLOW TRANSIENTS	ANL	6600	400
F4	RS	P	T	F$	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS	AI	360	309
F4	RS	P	T	F$	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS	AI	7094	309
F4	RS	P	T	F$	WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM	GGA	1108	R274
F4	RS	P	T	G$	AISITE2, PARAMETRIC SITE REQUIREMENT STUDY	AI	360	172
F4	RS	P	T	G$	ECCSA1, LOSS-OF-COOLANT + EMERGENCY COOLING	BCL	6400	330
F4	RS	P	T	G$	ELBOW, FUEL ELEMENT STRESS ANALYSIS STUDY	GGA	7044	229
F4	RS	P	T	G$	HAA3, COAGULATION OF HETEROGENEOUS AEROSOLS	AI	360	443
F4	RS	P	T	G$	NURLOC-1.0, LOSS-OF-COOLANT THERMAL ANALYSIS	BCL	6400	328
F4	RS	P	T	G$	RELAP2, REACTOR BLOWDOWN - EXCURSION ANALYSIS	INC	7044	369
F4	RS	P	T	H$	COBRA2, ROD BUNDLE THERMALHYDRAULIC ANALYSIS	BNWL	1108	432
F4	RS	P	T	H$	COBRA2, ROD BUNDLE THERMALHYDRAULIC ANALYSIS	AJ-ANL	360	432
F4	RS	P	T	H$	DYNAM, DYNAMIC ANALYSIS BOILING FLOW STEAM	GGA	1108	440
F4	RS	P	T	H$	MANTA, STEADY-STATE THERMAL-HYDRAULIC ANALYSIS	NED	635	256
F4	RS	P	T	I$	GEM, EIGENVALUE PROBLEM FOR VIBRATING SYSTEMS	KAPL	6600	R344
F4	RS	P	T	I$	SABOR4, DISCRETE-ELEMENT ANALYSIS THIN SHELLS	MIT	360	R402
F4	RS	P	T	I$	SAFE-AXISYM, STRESS ANALYSIS AXISYMMETRIC LOAD	GGA	7044	251
F4	RS	P	T	I$	SAFE-PORS, STRESS ANALYSIS AXISYMMETRIC LOAD	GGA	7044	250
F4	RS	P	T	I$	SAFE-3D, 3-D COMPOSITE STRUCTURE STRESS STUDY	GGA	1108	332
F4	RS	P	T	I$	SOR2, STRESS ANALYSIS SHELLS OF REVOLUTION	KAPL	6600	R 80
F4	RS	P	T	J$	MUSCAT, VIEW FACTOR SHIELDING CODE CAVITY GEOM	GGA	1108	259
F4	RS	P	T	L$	GANO, GAFGAR X-SECTION LIBRARY PREPARATION	GGA	1108	345
F4	RS	P	T	O$	BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS	GEV	635	311
F4	RS	P	T	O$	EXPN, ANALYSIS OF PULSED NEUTRON SOURCE DATA	NED	635	258
F4	RS	P	T	O$	JITER, FLUCTUATION EXPERIMENT ANALYSIS	BAPL	6600	R394
F4	RS	P	T	O$	TOAD, PROCESSING OF ANALYZER GAMMA-RAY SPECTRA	GGA	1108	333
F4	RS	P	X	A$	COHBE, COHERENT INELASTIC SCATTERING LAW CALC	GGA	1108	385
F4	RS	P	X	A$	LEGCOEF3, LEGENDRE COEF CALC FOR ANGULAR DIST	GGA	7044	217
F4	RS	P	X	G$	BURST1, HYDRODYNAMIC ANALYSIS DURING BLOWDOWN	PPCO	360	435
F4	RS	P	X	I$	FINEL, FINITE-ELEMENT STUDY 2,3-D STRUCTURES	KAPL	6600	R404
F4	RS	P	XT	A$	PUN1, UNRESOLVED RESONANCE INTEGRALS X-SECS	BAPL	6600	R359
F4	RS	P	XT	A$	SUMOR, S-WAVE NEUTRON X-SECTION CALCULATION	BAPL	6600	R399
F4	RS	P	XT	B$	GAF,GAR, SPECTRA AND GROUP-AVERAGED X-SEC CALC	GGA	1108	316
F4	RS	P	XT	B$	GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV	BC	625	185
F4	RS	P	XT	B$	HWCCR-SAFE, 2-D MONTE CARLO CELL CALCULATION	AI	360	307
F4	RS	P	XT	B$	PAX02, HARMONY-PDQ X-SECTION GENERATION CODE	BAPL	6600	R426
F4	RS	P	XT	B$	RESQ2,RESQO,DBF1, RESONANCE INTEGRAL HEX CELL	BAPL	6600	R285
F4	RS	P	XT	B$	1DX, 1-D DIFFUSION FAST X-SECTION GENERATION	BNW	1108	374
F4	RS	P	XT	C$	BE21, FEW-GP DISCRETE ORDINATES SLAB GEOMETRY	BAPL	6600	R398
F4	RS	P	XT	C$	VARI-QUIR3, 2-D MULTI-GP DIFFUSION XY RZ RTH	WANL	7094	264
F4	RS	P	XT	F$	CEXE,INCEXE, 1-GP 3-D XYZ XENON OSCILLATION	CE	360	415
F4	RS	P	XT	F$	FORE2, FAST REACTOR EXCURSION CALCULATIONS	ANL	360	174
F4	RS	P	XT	F$	FORE2, FAST REACTOR EXCURSION CALCULATIONS	NED	635	174
F4	RS	P	XT	F$	NOWIG, 1-D 2-GP KINETICS TEMPERATURE FEEDBACK	BAPL	6600	R371
F4	RS	P	XT	F$	TWIGL, 2-D 2-GP SPACE-TIME DIFFUSION FEEDBACK	BAPL	6600	R338
F4	RS	P	XT	F$	WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM	BAPL	6600	R274
F4	RS	P	XT	G$	FLASH3, LOSS-OF-COOLANT ACCIDENT ANALYSIS	BAPL	6600	R295
F4	RS	P	XT	G$	FLASH4, FULLY-IMPLICIT TRANSIENT SIMULATION	BAPL	6600	R448
F4	RS	P	XT	G$	MO555,ACT1, LOSS-OF-COOLANT ACCIDENT ANALYSIS	BAPL	6600	R284
F4	RS	P	XT	H$	FIGRO, LSBR FUEL SWELLING TEMPERATURE STUDY	BAPL	6600	R272
F4	RS	P	XT	H$	FLOT1,MO219, PWR FLOW TRANSIENT ANALYSIS	BAPL	6600	R331
F4	RS	P	XT	H$	GLUB1, WATER-LOGGED FUEL ELEMENT ANALYSIS	BAPL	6600	R424
F4	RS	P	XT	H$	HOT2, 2-D TRANSIENT HEAT CONDUCTION PROGRAM	BAPL	6600	R286
F4	RS	P	XT	H$	MO899,HOH, STEAM TABLES 14.5-2538 PSIA	BAPL	6600	R294

F4	RS P XT H$	TOPS, TRANSIENT THERMODYNAMICS OF PRESSURIZERS	BAPL	6600	R348
F4	RS P XT H$	WASP, WATER AND STEAM THERMODYNAMIC PROPERTIES	BAPL	6600	R396
F4	RS P XT I$	CYGRO3, OXIDE FUEL ROD STRESS & DEFORMATION	BAPL	6600	R449
F4	RS P XT I$	MANE1, RECTANGULAR MAGNETIC NETWORK SOLUTION	BAPL	6600	R412
F4	RS P XT I$	MO266, LINEAR ELASTIC STRUCTURAL DYNAMICS	BAPL	6600	R383
F4	RS P XT I$	MO457,PIPE, ELASTIC STRESS OF PIPING SYSTEM	BAPL	6600	R329
F4	RS P XT I$	MO552, DYNAMIC ANALYSIS LINEAR ELASTIC SYSTEMS	BAPL	6600	R283
F4	RS P XT I$	SEALSHELL2, SHELL STRESS ANALYSIS AXISYM LOAD	BAPL	6600	R282
F4	RS P XT J$	MO756,LETO, 1-D SLAB GAMMA-RAY TRANSPORT	BAPL	6600	R343
F4	RS P XT K$	CITATION, 1,2,3-D DIFFUSION DEPLETION MULTIGP	ORNL	360	387
F4	RS P XT L$	BL47, DRAFTING TOOL TO PLOT PLANE STRUCTURES	BAPL	6600	R373
F4	RS P XT O$	REDUX, REACTOR FLUCTUATION EXPERIMENT ANALYSIS	BAPL	6600	R425
F4	RS P XT P$	DAFT1, LEAST SQUARES FIT FISSILE NUCLIDE DATA	BAPL	6600	R327
F4	RS P XT P$	MO661,MO657,MO626, POLYNOMIAL CURVE FITTING	BAPL	6600	R411
F4	RS PL T B$	ETOG1, ENDF/B TO MUFT, GAM, ANISN X-SEC FORMAT	WNES	6600	437
F4	RS PL T B$	GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV	BNW	1107	185
F4	RS PL T B$	TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS	BHSC	360	50
F4	RS PL T C$	BISYN, 2-D MULTI-GP DIFFUSION SYNTHESIS CALC	NED	635	287
F4	RS PLXT B$	AILMOE, X-SECTION CALC ELASTIC SCAT RESONANCES	AI	360	147
F4	RS PLXT B$	GGC4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA	GGA	1108	298
F4	RS PLXT B$	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM	BNL-DP	360	277
F4	RS PLXT B$	LECFARC, SPECTRA CALCULATION WITH DEPLETION	WAPD	360	279
F4	RS PLXT B$	XSDRN, DISCRETE ORDINATE MULTIGROUP CONSTANTS	ORNL	360	393
F4	RS PLXT C$	MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	PURD	6500	262
F4	RSB L T M$	CLIP, FORM OR THREDES LIBRARY UTILITY ROUTINE	AI	360	271
F4	RSBP A$	MUFFLE, FISSILE NUCLIDE X-SECTION EVALUATION	ORNL	7090	323
F4	RSBP D$	CHAINS, ANALYSIS OF RADIOACTIVE DECAY CHAINS	GEC	635	418
F4	RSBP D$	DOS, NEUTRON FLUX-DOSIMETER ACTIVITY RELATION	GEC	635	423
F4	RSBP O$	MICHRD, MICROHARDNESS MEASUREMENT ANALYSIS	GEC	635	421
F4	RSBP T B$	GROUSE, SPACE-DEPENDENT X-SECTION GENERATION	GEC	635	420
F4	RSBP T D$	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	ANL	3600	325
F4	RSBP T K$	THREDES, 1-D FEW-GP DIFFUSION DESIGN SYSTEM	AI	360	273
F4	RSBP T N$	CROSSPLOT, SC4020 PLOTS FROM X-SECTION TAPES	GGA	7044	207
F4	RSBP XT D$	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	AI	360	58
F4	RSBPLXT C$	ANISN, 1-D MULTI-GP DISCRETE ORDINATE CALC	AI	360	151
F4	RSBPLXT C$	CAESAR4,LIBLST, 1-D MULTI-GP DIFFUSION + LIB	AI	360	270
F4	RSBPLXT D$	NAP, NEUTRON-INDUCED GAMMA-RAY RADIOACTIVITY	IITR	7094	314
F5	RS P A$	FLANGE1, SCATTERING LAW X-SECTION CALCULATION	GGA	1108	247
F5	RS P H$	FLAC, STEADY-STATE FLOW, PRESSURE DISTRIBUTION	GGA	1108	395
F5	RS P T H$	TAC2D, STEADY-STATE AND TRANSIENT TEMP CALC	GGA	1108	408
F5	RS P T H$	TAC3D, TRANSIENT 3-D HEAT TRANSFER PROGRAM	GGA	1108	414
F5	RSBPL T D$	ISOGEN, RADIONUCLIDE GENERATION AND DECAY	BNW	1108	367
F63	RS C$	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES	CDC	1604	30
F63	RS C$	TDP, 2-D PERTURBATION TOC OR 2DXY FLUX INPUT	PW	1604	199
F63	RS C$	2DF, 2-D MULTI-GP DISCRETE ORDINATE CODE	UNC-LASL	1604	173
F63	RS I$	WOPEXPRT, REACTOR WEIGHT OPTIMIZATION STUDY	PW	1604	190
F63	RS L$	CSP1, SN X-SECTION LIBRARY TAPE PREPARATION	PW	1604	194
F63	RS L$	CSP2A, SN X-SECTION LIBRARY TAPE PREPARATION	PW	1604	193
F63	RS T C$	2DXYL, 3-D MULTI-GP FLUX SYNTHESIS PROGRAM CYL	PW	1604	192
F63	RS L T D$	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	CDC	1604	58
F63	RS LXT B$	GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC	ANL	3600	33
F63	RS P A$	HAFEVER, HAUSER-FESHBACH INELASTIC SCATTERING	CDC	1604	14
F63	RS P A$	PEGGY, ELASTIC SCATTERING PHASE-SHIFT ANALYSIS	ORNL	1604	334
F63	RS P A$	RAMES, PARTICLE WAVE FUNCTION RADIAL INTEGRALS	ORNL	1604	335
F63	RS P B$	SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY	CDC	1604	52
F63	RS P B$	S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION	CDC	1604	53
F63	RS P D$	ISOCRUNCH, REACTION DECAY CHAIN ANALYSIS	ORNL	1604	180
F63	RS P D$	ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC	ORNL	1604	322
F63	RS P D$	ISOTOPES, MAXIMUM YIELD FROM REACTION OR DECAY	ORNL	1604	179
F63	RS P D$	POWERCO, NUCLEAR STATION ELECTRICITY COSTS	ORNL	1604	340
F63	RS P E$	AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK	CDC	1604	121
F63	RS P E$	CMPXMAT, TRANSFER FUNCTION EVALUATION	PW	1604	188
F63	RS P F$	AX-TNT, COUPLED NEUTRONICS-HYDRODYNAMICS SPH	PW	1604	191
F63	RS P G$	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	CDC	1604	47

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F63   RS P    H$ MO899,HOH, STEAM TABLES 14.5-2538 PSIA          ANL  1604  R294
F63   RS P    J$ GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY    CDC  1604    45
F63   RS P    J$ GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM  CDC  1604    46
F63   RS P    L$ SNC, CALCULATION OF SN CONSTANTS FOR DSN TDC   PW   1604   189
F63   RS P  T A$ JUPITOR1, COUPLED-CHANNEL X-SEC EVALUATION     ORNL 1604   308
F63   RS P  T C$ EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER SLAB  CDC  1604    87
F63   RS P  T C$ FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE CDC  1604    28
F63   RS P  T C$ FORTRAN TDC, 2-D MULTI-GP SN APPROXIMATION RZ  PW   1604   161
F63   RS P  T C$ WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY  CDC  1604    32
F63   RS P  T C$ 2OGRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER CDC  1604    40
F63   RS P  T I$ PIP, CENTRIFUGAL PUMP IMPELLER DESIGN STUDY    PW   1604   187
F63   RS PL T A$ ARES2, RESONANCE INTEGRAL X-SECTION CALC       CDC  1604    89
F63   RS PL T B$ FORM, FAST NEUTRON SPECTRUM X-SECTION CALC     CDC  1604    51
F63   RS PL T B$ TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS  CDC  1604    50
F63   RS PL T C$ AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE   CDC  1604    29
F63   RS PL T C$ DTF, 1-D MULTIGROUP DISCRETE ORDINATE CALC     UNC  1604   144
F63   RS PLXT B$ GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC     CDC  1604    33
F63   RS PLXT C$ FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE   CDC  1604   120
F63   RSBP    A$ NEARREX, COMPOUND NUCLEUS X-SECTION CALC       ANL  3600   171
F63   RSBP    E$ R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS   ANL  3600   255
F63   RSBP    E$ R102, SPACE-INDEPENDENT INVERSE KINETICS CALC  ANL  3600   168
F63   RSBP    F$ AX1, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE   CDC  3600   102
F63   RSBP    O$ COINC, COINCIDENCE COUNTING DATA REDUCTION     ANL  3600   248
F63   RSBP  T A$ 2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS  ANL  3600   254
F63   RSBP  T C$ FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION   CDC  3600   167
F63   RSBP  T E$ TRAFICORPORATION, TRANSFER FUNCTION SYNTHESIS  ANL  3600   135
F63   RSBP  T G$ R153, PARAMETRIC SITE REQUIREMENT STUDY        ANL  3600   172
F63   RSBP  T H$ ARGUS, TRANSIENT TEMPERATURE CALC CYLINDER     ANL  3600   152
F63   RSBP  T M$ XLIBIT, X-SECTION LIBRARY UTILITY ROUTINE      ANL  3600   181
F63   RSBP XT B$ THERMOS(ANL), THERMAL SPECTRUM X-SECTION CALC  ANL  3600   184
F63   RSBPL T J$ MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY    ANL  3600   143
GADOSE,DOSET, HTGR ACCIDENT ANALYSIS DOSE CALC GGA  1108 F4    RS    T G $  261
GAF,GAR, SPECTRA AND GROUP-AVERAGED X-SEC CALC GGA  1108 F4    RS P XT B $  316
GAFFE, EQUILIBRIUM FUEL CYCLE CALCULATION      GGA  1108 F4     S    T D $  302
GAFGAR ENDF/B DATA TAPES   GGA  11C8 BIN   R   L T Z$ ZPR-III ASSEMBLY 48   356
GAFGAR X-SECTION LIBRARY PREPARATION     GGA  1108 F4    RS P  T L$ GAND,   345
GAKER, INELASTIC SCAT X-SECTION CALC MODERATOR GGA  1108 F4    RS P    A $  289
GAKIN, 1-D MULTIGROUP TIME-DEPENDENT DIFFUSION GGA  1108 F4    RS P  T F $  310
GAKIT, 1-D MULTIGP KINETICS WITH TEMP FEEDBACK GGA  1108 F4    RS P  T F $  370
GAM, ANISN X-SEC FORMAT WNES 6600 F4     RS PL T B$ ETOG1, ENDF/B TO MUFT,   437
GAMBLE4, 2-D MULTI-GP DIFFUSION XY RZ GEOMETRY GGA  7044 F+MAP RSBP  T C $  222
GAMBLE5, 2-D MULTI-GP DIFFUSION XY RZ GEOMETRY GGA  1108 F+BAL RS P  T C $  222
GAMMA RAY SOURCE BUILDUP FACTOR CALC     BAPL 6600 F+ASC RS P XT J$ ASPIS,  R429
GAMMA RAY SPECTRA        CRNL  360 F4    RS P    O$ ALPHA-M, RESOLUTION OF   413
GAMMA-P, PRODUCTION X-SECTIONS FOR GAMMA-RAYS  GEC  7094 F4    RS P    B $  235
GAMMA-RAY ATTENUATION CYL SPHERE GEOM  AI   7090 F2    RS P    J$ GRACE2,    46
GAMMA-RAY ATTENUATION CYL SPHERE GEOM  CDC  1604 F63   RS P    J$ GRACE2,    46
GAMMA-RAY ATTENUATION SLAB GEOMETRY    AI   7090 F2    RS P    J$ GRACE1,    45
GAMMA-RAY ATTENUATION SLAB GEOMETRY    CDC  1604 F63   RS P    J$ GRACE1,    45
GAMMA-RAY DOSE RATE FROM A CLOUD         AI   7090 F2    RS P    G$ CLOUD,     47
GAMMA-RAY DOSE RATE FROM A CLOUD         CDC  1604 F63   RS P    G$ CLOUD,     47
GAMMA-RAY DOSE RATE FROM A CLOUD         DP    360 F4    RS P    G$ CLOUD,     47
GAMMA-RAY HISTORIES   PW   1604 F+CDP RS P XT B$ FMC-N,FMC-G, MC NEUTRON,    195
GAMMA-RAY PHOTOFRACTION SOLID CRYSTAL    UM   7090 MAD   RSB    O$ BURP4,     169
GAMMA-RAY PHOTOFRACTION WELL CRYSTAL     UM   7090 MAD   RSB    O$ BURP5,     170
GAMMA-RAY RADIOACTIVITY   IITR 7094 F4     RSBPLXT O$ NAP, NEUTRON-INDUCED    314
GAMMA-RAY SPECTRA GGA  1108 F4    RS P  T O$ TOAD, PROCESSING OF ANALYZER    333
GAMMA-RAY SPECTRA KSUN  360 F4    RS P    O$ CORGAM, UNFOLDING OF COMPLEX    390
GAMMA-RAY TRANSPORT        BAPL 6600 F4    RS P XT J$ MO756,LETO, 1-D SLAB  R343
GAMMA-RAYS  GEC  7094 F4    RS P    B$ GAMMA-P, PRODUCTION X-SECTIONS FOR   235
GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV    BC    625 F4    RS P XT B $  185
GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV    BNW  1107 F4    RS PL T B $  185
GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV    BNW  7090 F+FAP RS PL T B $  185
GAMTRI, 2-D MULTIGP DIFFUSION TRIANGULAR MESH  GGA  1108 F+BAL RS P  T C $  401
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GAM1 + GAM2 WNES 6600 BCD   R   L T Z$ ETOG1 DATA LIBRARIES, MUFT4 OR 5 +   447
GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC      ANL  3600 F63   RS  LXT B $   33
GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC      CDC  1604 F63   RS PLXT B $   33
GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC      GGA  7090 F2    RS PLXT B $   33
GAM2 WNES 6600 BCD   R   L T Z$ ETOG1 DATA LIBRARIES, MUFT4 OR 5 + GAM1 +   447
GAND, GAFGAR X-SECTION LIBRARY PREPARATION      GGA  1108 F4    RS P  T L $  345
GANDY, UNRESOLVED RESONANCE X-SECTION CALC      GGA  1108 F4    RS P    A $  341
GAPL3, INELASTIC LARGE DEFLECTION STRESS STUDY  BAPL 6600 F+COM RS P XT I $ R397
GAPOTKIN, SPACE-INDEPENDENT REACTOR KINETICS    GGA  1108 F4    RS P    E $  317
GAR, SPECTRA AND GROUP-AVERAGED X-SEC CALC GGA  1108 F4    RS P XT B$ GAF,  316
GARGCYLE, FUEL CYCLE ANALYSIS PARTIAL REFUEL    GGA  7044 F4    RS P  T D $  260
GARCL, RESONANCE OVERLAP AND LATTICE EFFECTS    GGA  7044 F+MAP RS P  T B $  219
GAROL, RESONANCE OVERLAP AND LATTICE EFFECTS    LER  7094 F4    RS P  T B $  219
GAS-COOLED CORE THERMAL DESIGN STUDY GGA  7044 F4    RS P    I$ GAZELLE5,   232
GASA, STABILITY ANALYSIS REACTOR KINETICS EQNS GGA  1108 F4    RS P    E $  290
GASKET, THERMAL SCATTERING LAW CALCULATION      GGA  1108 F4    RS P    A $  263
GASP2, 1-D FEW-GP DIFFUSION POWER DIST SEARCH   GGA  7044 F4    RS P  T C $  220
GASP7, 1-D BURNUP POWER DISTRIBUTION SEARCH     GGA  1108 F4    RS P  T C $  319
GATT, 3-D FEW-GP DIFFUSION CALC HEX-Z MESH      GGA  1108 F4    RS P  T C $  380
GAUGE, 2-D FEW-GP HEX GEOM DIFFUSION DEPLETION GGA  1108 F4    RS P  T D $  339
GAVER, ENERGY INTERVAL X-SECTION AVERAGING      GGA  7044 F4    RS P    L $  218
GAZELLE5, GAS-COOLED CORE THERMAL DESIGN STUDY GGA  7044 F4    RS P    I $  232
GAZE2, 1-D MULTIGROUP DIFFUSION SLAB, SPH, CYL GGA  1108 F4    RS P  T C $  430
GE-HAPO-S13, 1-D MULTI-GP DOUBLE SN APPROX      BNW  7090 FLOCO RSBP  T C $   75
GEC   625 F4    RS P    B$ AVOID, ANNULAR VOID X-SECTION CALCULATION          276
GEC   635 F+FAP RSBP  T Q$ CASCADE,CLUSTER, RADIATION DAMAGE IN METALS        419
GEC   635 F+GMP RSBP  T H$ THTE, 3-D TRANSIENT HEAT TRANSFER PROGRAM          346
GEC   635 F4    RSBP    D$ CHAINS, ANALYSIS OF RADIOACTIVE DECAY CHAINS       418
GEC   635 F4    RSBP    C$ DOS, NEUTRON FLUX-DOSIMETER ACTIVITY RELATION      423
GEC   635 F4    RSBP    C$ MICHRD, MICROHARDNESS MEASUREMENT ANALYSIS         421
GEC   635 F4    RSBP  T B$ GROUSE, SPACE-DEPENDENT X-SECTION GENERATION       420
GEC  7090 FAP   RSBP XT Q$ SPECTRA, MC CALCULATION IRRADIATED MATERIAL        422
GEC  7094 F4    RS P    B$ GAMMA-P, PRODUCTION X-SECTIONS FOR GAMMA-RAYS      235
GEM, EIGENVALUE PROBLEM FOR VIBRATING SYSTEMS   KAPL 6600 F4    RS P  T I $ R344
GENERATOR DESIGN CRITERIA COSTS  GGA  7044 F4    RS P  T D$ STMGEN, STEAM     227
GEOMETRICAL DATA HEAT TRANSFER STUDY SLL  6600 F4    RS P    N$ HEATMESH,    434
GEV   635 F4    RS P  T C$ BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS     311
GEV  7094 F+FAP RS P  T B$ EPITHERMCS, SPECTRUM AND X-SECTION CALCULATION     201
GGA  1108 BIN   R   L T Z$ ZPR-III ASSEMBLY 48 GAFGAR ENDF/B DATA TAPES       356
GGA  1108 F+BAL RS P  T C$ GAMBLE5, 2-D MULTI-GP DIFFUSION XY RZ GEOMETRY     222
GGA  1108 F+BAL RS P  T C$ GAMTRI, 2-D MULTIGP DIFFUSION TRIANGULAR MESH      401
GGA  1108 F4     S    T D$ GAFFE, EQUILIBRIUM FUEL CYCLE CALCULATION         302
GGA  1108 F4    RS      B$ HEXSCAT, ELASTIC SCAT X-SECTIONS HEX LATTICE       291
GGA  1108 F4    RS    T A$ FLANGE2, ENDF/B THERMAL SCATTERING DATA PROC       368
GGA  1108 F4    RS    T G$ GADOSE,DOSET, HTGR ACCIDENT ANALYSIS DOSE CALC     261
GGA  1108 F4    RS P    A$ GAKER, INELASTIC SCAT X-SECTION CALC MODERATOR     289
GGA  1108 F4    RS P    A$ GANDY, UNRESOLVED RESONANCE X-SECTION CALC         341
GGA  1108 F4    RS P    A$ GASKET, THERMAL SCATTERING LAW CALCULATION         263
GGA  1108 F4    RS P    A$ PSEUDO, STATISTICAL RESONANCE PARAMETER CALC       292
GGA  1108 F4    RS P    C$ FREVAP6, HTGR METALLIC FISSION PRODUCT RELEASE     301
GGA  1108 F4    RS P    E$ GAPOTKIN, SPACE-INDEPENDENT REACTOR KINETICS       317
GGA  1108 F4    RS P    E$ GASA, STABILITY ANALYSIS REACTOR KINETICS EQNS     290
GGA  1108 F4    RS P    I$ SAFE-CREEP, VISCOELASTIC ANALYSIS CONCRETE         300
GGA  1108 F4    RS P    I$ SAFE-PLANE, PLANE STRESS ANALYSIS, 2-D BODIES      252
GGA  1108 F4    RS P    I$ SAFE-SHELL, STRESS ANALYSIS THIN SHELLS            253
GGA  1108 F4    RS P    I$ SAFE-2D, PLANE + AXISYMMETRIC STRESS ANALYSIS      379
GGA  1108 F4    RS P    L$ GRDWRK, GRID GENERATION FOR SAFE PROGRAMS          296
GGA  1108 F4    RS P  T A$ TACASI, ANALYSIS OF RESONANCE MEASUREMENTS         410
GGA  1108 F4    RS P  T C$ GASP7, 1-D BURNUP POWER DISTRIBUTION SEARCH        319
GGA  1108 F4    RS P  T C$ GATT, 3-D FEW-GP DIFFUSION CALC HEX-Z MESH         380
GGA  1108 F4    RS P  T C$ GAZE2, 1-D MULTIGROUP DIFFUSION SLAB, SPH, CYL     430
GGA  1108 F4    RS P  T C$ TEMCO7, TEMPERATURE COEFFICIENT CALCULATION        320
GGA  1108 F4    RS P  T D$ BUGTRI, 2-D MULTIGP DIFFUSION+BURNUP TRI-MESH      439
GGA  1108 F4    RS P  T D$ BUG2, 2-D MULTIGROUP DIFFUSION & BURNUP XY, RZ     438
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GGA  1108 F4     RS P  T C$ FEVER7, 1-D MULTIGRCUP DIFFUSION AND DEPLETION   318
GGA  1108 F4     RS P  T C$ GAUGE, 2-D FEW-GP HEX GEOM DIFFUSION DEPLETION   339
GGA  1108 F4     RS P  T C$ PWCOST, REACTOR FUEL CYCLE COST CALCULATION      441
GGA  1108 F4     RS P  T E$ BLOOST6, CCMBINED KINETICS 2-D HEAT TRANSFER     303
GGA  1108 F4     RS P  T F$ GAKIN, 1-D MULTIGROUP TIME-DEPENDENT DIFFUSION   310
GGA  1108 F4     RS P  T F$ GAKIT, 1-D MULTIGP KINETICS WITH TEMP FEEDBACK   370
GGA  1108 F4     RS P  T F$ WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM     R274
GGA  1108 F4     RS P  T H$ DYNAM, DYNAMIC ANALYSIS BOILING FLOW STEAM       440
GGA  1108 F4     RS P  T I$ SAFE-3D, 3-D COMPOSITE STRUCTURE STRESS STUDY    332
GGA  1108 F4     RS P  T J$ MUSCAT, VIEW FACTOR SHIELDING CODE CAVITY GEOM   259
GGA  1108 F4     RS P  T L$ GAND, GAFGAR X-SECTICN LIBRARY PREPARATION       345
GGA  1108 F4     RS P  T C$ TOAD, PROCESSING OF ANALYZER GAMMA-RAY SPECTRA   333
GGA  1108 F4     RS P X  A$ COHBE, COHERENT INELASTIC SCATTERING LAW CALC    385
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1108 F4     RS P  T C$ GASP7, 1-D BURNUP POWER DISTRIBUTION SEARCH    GGA    319
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1108 F4     RS P  T C$ TEMCO7, TEMPERATURE COEFFICIENT CALCULATION    GGA    320
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1108 F4     RS P  T D$ BUGTRI, 2-D MULTIGP DIFFUSION+BURNUP TRI-MESH  GGA    439
1108 F4     RS P  T D$ BUG2, 2-D MULTIGROUP DIFFUSION & BURNUP XY, RZ GGA    438
1108 F4     RS P  T D$ FEVER7, 1-D MULTIGROUP DIFFUSION AND DEPLETION GGA    318
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1108 F4     RS P  T D$ GAUGE, 2-D FEW-GP HEX GEOM DIFFUSION DEPLETION GGA     339
1108 F4     RS P  T C$ PWCOST, REACTOR FUEL CYCLE COST CALCULATION     GGA     441
1108 F4     RS P  T D$ 2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION     BNW     325
1108 F4     RS P  T E$ BLCOST6, COMBINED KINETICS 2-D HEAT TRANSFER    GGA     303
1108 F4     RS P  T F$ GAKIN, 1-D MULTIGROUP TIME-DEPENDENT DIFFUSION GGA     310
1108 F4     RS P  T F$ GAKIT, 1-D MULTIGP KINETICS WITH TEMP FEEDBACK GGA     370
1108 F4     RS P  T F$ WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM     GGA    R274
1108 F4     RS P  T H$ COBRA2, ROD BUNDLE THERMALHYDRAULIC ANALYSIS    BNWL    432
1108 F4     RS P  T H$ DYNAM, DYNAMIC ANALYSIS BOILING FLOW STEAM      GGA     440
1108 F4     RS P  T I$ SAFE-3D, 3-D COMPOSITE STRUCTURE STRESS STUDY   GGA     332
1108 F4     RS P  T J$ MUSCAT, VIEW FACTOR SHIELDING CODE CAVITY GEOM  GGA     259
1108 F4     RS P  T L$ GAND, GAFGAR X-SECTION LIBRARY PREPARATION      GGA     345
1108 F4     RS P  T C$ TOAD, PROCESSING OF ANALYZER GAMMA-RAY SPECTRA GGA     333
1108 F4     RS P X  A$ COHBE, COHERENT INELASTIC SCATTERING LAW CALC   GGA     385
1108 F4     RS P XT B$ GAF,GAR, SPECTRA AND GROUP-AVERAGED X-SEC CALC  GGA     316
1108 F4     RS P XT B$ 1DX, 1-D DIFFUSION FAST X-SECTION GENERATION    BNW     374
1108 F4     RS PLXT B$ GOC4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA  GGA     298
1108 F5     RS P    A$ FLANGE1, SCATTERING LAW X-SECTION CALCULATION   GGA     247
1108 F5     RS P    H$ FLAC, STEADY-STATE FLOW, PRESSURE DISTRIBUTION  GGA     395
1108 F5     RS P  T H$ TAC2D, STEADY-STATE AND TRANSIENT TEMP CALC     GGA     408
1108 F5     RS P  T H$ TAC3D, TRANSIENT 3-D HEAT TRANSFER PROGRAM      GGA     414
1108 F5     RSBPL T D$ ISOGEN, RADIONUCLIDE GENERATION AND DECAY       BNW     367
14.5-14,500PSIA 32-472DEGF BAPL 6600 F4     RS    XT H$ WATER, STEAM TABLES R267
14.5-2538 PSIA          ANL  1604 F63    RS P     H$ MO899,HOH, STEAM TABLES R294
14.5-2538 PSIA          BAPL 6600 F4     RS P XT H$ MO899,HOH, STEAM TABLES R294
1604 F+CDP RS P  T P$ LAG, ASSEMBLER FOR FLOCO2 INSTRUCTION SET      PW      186
1604 F+CDP RS P XT B$ FMC-N,FMC-G, MC NEUTRON, GAMMA-RAY HISTORIES   PW      195
1604 F63    RS       C$ PERT, 1-D PERTURBATION FOR AIM AND FOG CODES CDC       30
1604 F63    RS       C$ TDP, 2-D PERTURBATION TDC OR 2DXY FLUX INPUT PW      199
1604 F63    RS       C$ 2DF, 2-D MULTI-GP DISCRETE ORDINATE CODE   UNC-LASL 173
1604 F63    RS       I$ WOPEXPRT, REACTOR WEIGHT OPTIMIZATION STUDY     PW      190
1604 F63    RS       L$ CSP1, SN X-SECTION LIBRARY TAPE PREPARATION     PW      194
1604 F63    RS       L$ CSP2A, SN X-SECTION LIBRARY TAPE PREPARATION    PW      193
1604 F63    RS     T C$ 2DXYL, 3-D MULTI-GP FLUX SYNTHESIS PROGRAM CYL PW      192
1604 F63    RS  L  T D$ SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION     CDC      58
1604 F63    RS P     A$ HAFEVER, HAUSER-FESHBACH INELASTIC SCATTERING CDC      14
1604 F63    RS P     A$ PEGGY, ELASTIC SCATTERING PHASE-SHIFT ANALYSIS ORNL    334
1604 F63    RS P     A$ RAMES, PARTICLE WAVE FUNCTION RADIAL INTEGRALS ORNL    335
1604 F63    RS P     B$ SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY  CDC      52
1604 F63    RS P     B$ S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION    CDC      53
1604 F63    RS P     D$ ISOCRUNCH, REACTION DECAY CHAIN ANALYSIS       ORNL    180
1604 F63    RS P     C$ ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC ORNL    322
1604 F63    RS P     D$ ISOTOPES, MAXIMUM YIELD FROM REACTION OR DECAY ORNL    179
1604 F63    RS P     C$ POWERCO, NUCLEAR STATION ELECTRICITY COSTS     ORNL    340
1604 F63    RS P     E$ AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK  CDC      121
1604 F63    RS P     E$ CMPXMAT, TRANSFER FUNCTION EVALUATION          PW      188
1604 F63    RS P     F$ AX-TNT, COUPLED NEUTRONICS-HYDRODYNAMICS SPH   PW      191
1604 F63    RS P     G$ CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD        CDC       47
1604 F63    RS P     H$ MO899,HOH, STEAM TABLES 14.5-2538 PSIA         ANL     R294
1604 F63    RS P     J$ GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY    CDC       45
1604 F63    RS P     J$ GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM  CDC       46
1604 F63    RS P     L$ SNC, CALCULATION OF SN CONSTANTS FOR DSN TDC   PW      189
1604 F63    RS P   T A$ JUPITOR1, COUPLED-CHANNEL X-SEC EVALUATION     ORNL    308
1604 F63    RS P   T C$ EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER SLAB  CDC       87
1604 F63    RS P   T C$ FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE CDC       28
1604 F63    RS P   T C$ FORTRAN TDC, 2-D MULTI-GP SN APPROXIMATION RZ  PW      161
1604 F63    RS P   T C$ WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY  CDC       32
1604 F63    RS P   T C$ 2DGRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER CDC       40
1604 F63    RS P   T I$ PIP, CENTRIFUGAL PUMP IMPELLER DESIGN STUDY    PW      187
1604 F63    RS PL  T A$ ARES2, RESONANCE INTEGRAL X-SECTION CALC       CDC       89
1604 F63    RS PL  T B$ FORM, FAST NEUTRON SPECTRUM X-SECTION CALC     CDC       51
1604 F63    RS PL  T B$ TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS  CDC       50
1604 F63    RS PL  T C$ AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE   CDC       29
1604 F63    RS PL  T C$ DTF, 1-D MULTIGROUP DISCRETE ORDINATE CALC     UNC      144
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1604 F63   RS PLXT B$ GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC      CDC     33
1604 F63   RS PLXT C$ FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE   CDC    120
1604 LAG1  RS P    C$ MGCSN, 1-D MULTI-GP SN APPROX SLAB CYL SPHERE  PW     211
1604 LAG1  RS P    N$ DTX, EFFECTIVE X-SECTION CALC FROM DSN OUTPUT  PW     210
2- OR 3-D HEAT CONDUCTION LUMPED MASS     LASL 7030 F4     RS    T H$ RATH,  242
2- OR 3-D HEAT CONDUCTION LUMPED MASS     LASL 7094 FAP    RS P  T H$ RATH,  242
2-D + 3-D PLOTTING   KAPL 6600 F+ASC RS P  T P$ DATATRAN UTILITY MODULES,  R407
2-D BODIES  GGA  1108 F4    RS P    I$ SAFE-PLANE, PLANE STRESS ANALYSIS,  252
2-D DESIGN PROGRAMS  LASL 7090 F2    RS P  T L$ DPC, DATA PREPARATION FOR  234
2-D DIFFUSION ABSORPTION REMOVAL X-SECS BAPL 6600 F4     RS   XT B$ MO807, R280
2-D EXCURSION CALCULATION R-Z GEOMETRY   APDA 7094 F4    RS P    F$ MARS,  293
2-D FEW-GP DIFFUSION BURNUP RZ GEOMETRY   GGA  7090 F+FAP RSBP  T D$ DDB,   99
2-D FEW-GP HEX GEOM DIFFUSION DEPLETION GGA  1108 F4    RS P  T D$ GAUGE,  339
2-D FEW-GP S4 APPROXIMATION RZ GEOMETRY  LRL   709 F2    RS      C$ FIRN,    7
2-D FEW-GROUP DIFFUSION AND DEPLETION    IBM   360 F+BAL RS P  T D$ PDQ5, R336
2-D FEW-GROUP DIFFUSION SLAB CYLINDER CDC  1604 F63   RS P  T C$ 20GRAND,   40
2-D FEW-GROUP DIFFUSION SLAB CYLINDER ORNL 7090 F2    RS P  T C$ 20GRAND,   40
2-D GEOMETRY INPUT, PREPARATION, EDIT KAPL 6600 F+ASC RS P  T L$ DATATRAN R406
2-D HEAT TRANSFER   GGA  1108 F4    RS P  T E$ BLOOST6, COMBINED KINETICS  303
2-D HEAT TRANSFER GGA  7044 F4    RS    T H$ BLOOST5, POINT-KINETICS WITH  205
2-D MONTE CARLO CELL CALCULATION   AI    360 F4    RS P XT B$ HWOCR-SAFE,  307
2-D MULTI-GP DIFFUSION     UMCC  360 F4    RS P  T C$ VARI-QUIR, TIME-DEP  212
2-D MULTI-GP DIFFUSION     WANL 6600 F4    RS P  T C$ VARI-QUIR, TIME-DEP  212
2-D MULTI-GP DIFFUSION DEPLETION CODE ORNL 7090 F+FAP RS P  T D$ ASSAULT,  240
2-D MULTI-GP DIFFUSION PROGRAM   AAEC  360 F+BAL RS P  T C$ CRAM, 1-D AND  103
2-D MULTI-GP DIFFUSION PROGRAM   ORNL 7090 F+FAP RS P  T C$ EXTERMINATOR,  156
2-D MULTI-GP DIFFUSION PROGRAM   UK-R 7090 F+FAP RSBPL T C$ CRAM, 1-D AND  103
2-D MULTI-GP DIFFUSION PROGRAM  BC    625 F4    RS    T C$ EXTERMINATOR2,  156
2-D MULTI-GP DIFFUSION PROGRAM  ORNL  360 F4    RS P  T C$ EXTERMINATOR2,  156
2-D MULTI-GP DIFFUSION SYNTHESIS CALC   NED   635 F4    RS PL T C$ BISYN,  287
2-D MULTI-GP DIFFUSION XY RZ GEOMETRY GGA  1108 F+BAL RS P  T C$ GAMBLE5,  222
2-D MULTI-GP DIFFUSION XY RZ GEOMETRY GGA  7044 F+MAP RSBP  T C$ GAMBLE4,  222
2-D MULTI-GP DIFFUSION XY RZ RTH   WANL 7094 F4    RS P XT C$ VARI-QUIR3,  264
2-D MULTI-GP DISCRETE ORDINATE CODE   UNC-LASL 1604 F63   RS      C$ 2DF,  173
2-D MULTI-GP SN APPROXIMATION RZ  PW   1604 F63   RS P  T C$ FORTRAN TDC,  161
2-D MULTI-GP SN APPROXIMATION XY GEOM    AGC  7090 FLOCO RSBP    C$ 2DXY,   18
2-D MULTI-GP TRANSPORT CODE R-Z GEOM  LASL 1108 F4    RS P  T C$ TWOTRAN,  358
2-D MULTI-GP TRANSPORT CODE X-Y GEOM  ANL   360 F4    RS P  T C$ TWOTRAN,  358
2-D MULTI-GP TRNSPT CODE XY RZ RTHETA LASL 6600 F4    RS P  T C$ TWOTRAN,  358
2-D MULTIGP DIFFUSION TRIANGULAR MESH  GGA  1108 F+BAL RS P  T C$ GAMTRI,  401
2-D MULTIGP DIFFUSION+BURNUP TRI-MESH  GGA  1108 F4    RS P  T D$ BUGTRI,  439
2-D MULTIGROUP DIFFUSION & BURNUP XY, RZ GGA  1108 F4    RS P  T D$ BUG2,  438
2-D MULTIGROUP DIFFUSION AND DEPLETION    ANL  3600 F4    RSBP  T D$ 2DB,  325
2-D MULTIGROUP DIFFUSION AND DEPLETION    BNW  1108 F4    RS P  T D$ 2DB,  325
2-D MULTIGROUP DIFFUSION AND DEPLETION    LASL 6600 F4    RS P  T D$ 2DB,  325
2-D MULTIGROUP DISCRETE ORDINATE PROGRAM LER  7090 F+MAP RS P  T C$ TDSN,  312
2-D PERTURBATION TDC OR 2DXY FLUX INPUT   PW   1604 F63   RS      C$ TDP,  199
2-D PERTURBATION XY RZ RTHETA GEOMETRY  BNW  1108 F4    RS P    C$ PERT4,  304
2-D TIME-DEPENDENT HEAT CONDUCTION      PPCO 7044 F+MAP RS P  T H$ TOCDEE,  349
2-D TRANSIENT HEAT CONDUCTION PROGRAM    BAPL 6600 F4    RS P XT H$ HOT2, R286
2-D 2-GP DIFFUSION CYLINDER SLAB  CDC  1604 F63   RS P  T C$ EQUIPOISE3A,   87
2-D 2-GP DIFFUSION CYLINDER SLAB  ORNL 7090 F2    RS P  T C$ EQUIPOISE3A,   87
2-D 2-GP SPACE-TIME DIFFUSION          ANL   360 F4    RS P    F$ TWIGGLE, R338
2-D 2-GP SPACE-TIME DIFFUSION FEEDBACK  BAPL 6600 F4    RS P XT F$ TWIGL, R338
2-D 2-GROUP DIFFUSION SLAB CYL      ORNL 7090 F2    RS P  T C$ EQUIPOISE3,   39
2-D 2-PHASE FLOW     AI   7094 F2    RS P    H$ FLOW-MODEL, MULTI-CHANNEL  246
2-GP DIFFUSION CYLINDER SLAB  CDC  1604 F63   RS P  T C$ EQUIPOISE3A, 2-D   87
2-GP DIFFUSION CYLINDER SLAB  ORNL 7090 F2    RS P  T C$ EQUIPOISE3A, 2-D   87
2-GP KINETICS TEMPERATURE FEEDBACK  BAPL 6600 F4    RS P XT F$ NOWIG, 1-D R371
2-GP SPACE-TIME DIFFUSION          ANL   360 F4    RS P    F$ TWIGGLE, 2-D R338
2-GP SPACE-TIME DIFFUSION FEEDBACK  BAPL 6600 F4    RS P XT F$ TWIGL, 2-D R338
2-GP SPACE-TIME DIFFUSION 3-GEOM    BAPL 6600 F4    RS P XT F$ WIGL2, 1-D R274
2-GP SPACE-TIME DIFFUSION 3-GEOM    GGA  1108 F4    RS P  T F$ WIGL2, 1-D R274
2-GROUP DIFFUSION SLAB CYL      ORNL 7090 F2    RS P  T C$ EQUIPOISE3, 2-D   39
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2-GROUP DIFFUSION XYZ GEOMETRY  CDC  1604 F63   RS P  T C$ WHIRLAWAY, 3-D     32
2-GROUP DIFFUSION XYZ GEOMETRY  ORNL 7090 F2    RS P  T C$ WHIRLAWAY, 3-D     32
2-PHASE FLOW     AI   7094 F2    RS P    H$ FLOW-MODEL, MULTI-CHANNEL 2-D   246
2-REG CELL  ANL  3600 F36   RSBP XT 6$ RIFF-RAFF, RESONANCE INTEGRAL CALC   213
2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION     ANL  3600 F4    RSBP  T D $  325
2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION     BNW  1108 F4    RS P  T D $  325
2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION     LASL 6600 F4    RS P  T D $  325
2DF, 2-D MULTI-GP DISCRETE ORDINATE CODE   UNC-LASL 1604 F63   RS       C $  173
2DXY FLUX INPUT   PW   1604 F63   RS       C$ TDP, 2-D PERTURBATION TOC OR   199
2DXY, 2-D MULTI-GP SN APPROXIMATION XY GEOM     AGC  7090 FLOCO RSBP     C $   18
2DXYL, 3-D MULTI-GP FLUX SYNTHESIS PROGRAM CYL PW   1604 F63   RS     T C $  192
2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS   AI   7094 F2    RS P  T A $  254
2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS   ANL  3600 F63   RSBP  T A $  254
2OGRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER CDC  1604 F63   RS P  T C $   40
2OGRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER ORNL 7090 F2    RS P  T C $   40
2000 F2     RS P     A$ DOPIE, RESOLVED RESONANCE X-SECTION CALC        NED    177
2000 F2     RS P     A$ RAPTURE, RESONANCE INTEGRAL X-SECTION CALC      NED    176
2000 F2     RS P     B$ SPARTA, SPATIALLY-AVERAGED DOPPLER EFFECTS      NED    178
2000 F4     RS       A$ STRIP, RESOLVED RESONANCE INTEGRAL CALCULATION BW      305
2000 F4     RS P  T F$ FORE, FAST REACTOR EXCURSION CALCULATIONS       NED    174
3-D COMPOSITE STRUCTURE STRESS STUDY  GGA  1108 F4    RS P  T I$ SAFE-3D,  332
3-D DIFFUSION DEPLETION MULTIGP  ORNL  360 F4    RS P XT K$ CITATION, 1,2,  387
3-D FEW-GP DIFFUSION CALC HEX-Z MESH     GGA  1108 F4    RS P  T C$ GATT,   380
3-D FEW-GP DIFFUSION DEPLETION     BAPL 6600 F+ASC RS P XT D$ PDQ7, 1,2 OR  R275
3-D FEW-GP DIFFUSION DEPLETION     IBM  360 F4    RS P  T D$ PDQ7, 1,2 OR  R275
3-D HEAT CONDUCTION LUMPED MASS    LASL 7030 F4    RS     T H$ RATH, 2- OR   242
3-D HEAT CONDUCTION LUMPED MASS    LASL 7094 FAP   RS P  T H$ RATH, 2- OR   242
3-D HEAT TRANSFER PROGRAM     GGA  1108 F5    RS P  T H$ TAC3D, TRANSIENT  414
3-D MULTI-GP FLUX SYNTHESIS PROGRAM CYL PW   1604 F63   RS     T C$ 2DXYL,  192
3-D PLOTTING   KAPL 6600 F+ASC RS P  T P$ DATATRAN UTILITY MODULES, 2-D +  R407
3-D REACTIVITY AND POWER DISTRIBUTION   CDC  3600 F63   RSBP  T C$ FLARE,   167
3-D REACTIVITY AND POWER DISTRIBUTION   NED  635 F4    RS P  T C$ FLARE,   167
3-D STRUCTURES   KAPL 6600 F4    RS P X  I$ FINEL, FINITE-ELEMENT STUDY 2, R404
3-D TEMPERATURE DISTRIBUTION PROGRAM     KAPL 6600 F+ASC RS P  T H$ LION,  R299
3-D TRANSIENT HEAT TRANSFER PROGRAM      GEC  635 F+GMP RSBP  T H$ THTE,   346
3-D XYZ XENON OSCILLATION    CE    360 F4    RS P XT F$ CEXE,INCEXE, 1-GP  415
3-D 2-GROUP DIFFUSION XYZ GEOMETRY  CDC  1604 F63   RS P  T C$ WHIRLAWAY,   32
3-D 2-GROUP DIFFUSION XYZ GEOMETRY  ORNL 7090 F2    RS P  T C$ WHIRLAWAY,   32
3-GEOM    BAPL 6600 F4    RS P XT F$ WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION  R274
3-GEOM    GGA  1108 F4    RS P  T F$ WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION  R274
32-472DEGF BAPL 6600 F4    RS   XT H$ WATER, STEAM TABLES 14.5-14,500PSIA R267
360 F+BAL RS  L T M$ SCORE2, SCISRS ENDF/B GRAPHIC X-SEC EVALUATION AI     375
360 F+BAL RS P  T B$ RAFFLE, 1ST FLIGHT COLLISION PROBABILITIES MC  ORNL  392
360 F+BAL RS P  T C$ CRAM, 1-D AND 2-D MULTI-GP DIFFUSION PROGRAM   AAEC  103
360 F+BAL RS P  T D$ PDQ5, 2-D FEW-GROUP DIFFUSION AND DEPLETION    IBM  R336
360 F+BAL RS P  T E$ AIROS2A, SIMULATION OF REACTOR DYNAMICS        AI    326
360 F+BAL RSBP  T G$ CONTEMPT,CONPS, PRESSURE-TEMPERATURE RESPONSE  INC   433
360 F4      S P    B$ WELWING, MATERIAL BUCKLING CYL FUEL ELEMENTS  AEB   362
360 F4     RS  L T C$ FIRE5, 1-D AGE-DIFFUSION SLAB CYLINDER SPHERE AEB     9
360 F4     RS P     A$ CODILLI, LEAST SQUARES ANALYSIS RESONANCE DATA UILL  347
360 F4     RS P     C$ PERT, 1-D PERTURBATION FOR AIM AND FOG CODES  BHSC   30
360 F4     RS P     D$ ISOSEARCH, ISOTOPE PRODUCTION FLUX, X-SEC CALC ORNL  322
360 F4     RS P     E$ AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK AEB   121
360 F4     RS P     E$ BLAST, REACTOR KINETICS TEMPERATURE DIST STUDY AEB   363
360 F4     RS P     F$ TWIGGLE, 2-D 2-GP SPACE-TIME DIFFUSION        ANL  R338
360 F4     RS P     G$ CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD       DP    47
360 F4     RS P     G$ WATER-HAMMER, LIQUID BLOWDOWN ANALYSIS LOFT   UGA   278
360 F4     RS P     O$ ALPHA-M, RESOLUTION OF GAMMA RAY SPECTRA      ORNL  413
360 F4     RS P     O$ CORGAM, UNFOLDING OF COMPLEX GAMMA-RAY SPECTRA KSUN 390
360 F4     RS P  T A$ ATHENA4, INELASTIC SCATTERING FORM FACTORS    ORNL  417
360 F4     RS P  T A$ FLANGE2, ENDF/B THERMAL SCATTERING DATA PROC  DP    368
360 F4     RS P  T B$ SUPERTOG, ENDF/B FINE-GP CONSTANTS GENERATION ORNL  431
360 F4     RS P  T C$ DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM  ANL   209
360 F4     RS P  T C$ EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PROGRAM ORNL  156
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360 F4     RS P  T C$ FAIMOS, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE BHSC     120
360 F4     RS P  T C$ TWOTRAN, 2-D MULTI-GP TRANSPORT CODE X-Y GEOM  ANL      358
360 F4     RS P  T C$ VARI-QUIR, TIME-DEP 2-D MULTI-GP DIFFUSION     UMCC     212
360 F4     RS P  T D$ CINCAS, NUCLEAR FUEL CYCLE COST AND ECONOMICS  COMM     354
360 F4     RS P  T D$ PDQ7, 1,2 OR 3-D FEW-GP DIFFUSION DEPLETION    IBM     R275
360 F4     RS P  T F$ SAS1A, FAST REACTOR POWER AND FLOW TRANSIENTS  ANL      400
360 F4     RS P  T F$ TSN, SPATIALLY-DEPENDENT REACTOR KINETICS      AI       309
360 F4     RS P  T G$ AISITE2, PARAMETRIC SITE REQUIREMENT STUDY     AI       172
360 F4     RS P  T G$ HAA3, COAGULATION OF HETEROGENEOUS AEROSOLS    AI       443
360 F4     RS P  T H$ COBRA2, ROD BUNDLE THERMALHYDRAULIC ANALYSIS AJ-ANL     432
360 F4     RS P  T I$ SABOR4, DISCRETE-ELEMENT ANALYSIS THIN SHELLS  MIT     R402
360 F4     RS P X  G$ BURST1, HYDRODYNAMIC ANALYSIS DURING BLOWDOWN  PPCO     435
360 F4     RS P XT B$ HWOCR-SAFE, 2-D MONTE CARLO CELL CALCULATION   AI       307
360 F4     RS P XT F$ CEXE,INCEXE, 1-GP 3-D XYZ XENON OSCILLATION    CE       415
360 F4     RS P XT F$ FORE2, FAST REACTOR EXCURSION CALCULATIONS     ANL      174
360 F4     RS P XT K$ CITATION, 1,2,3-D DIFFUSION DEPLETION MULTIGP  ORNL     387
360 F4     RS PL T B$ TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS  BHSC      50
360 F4     RS PLXT B$ AILMOE, X-SECTION CALC ELASTIC SCAT RESONANCES AI       147
360 F4     RS PLXT B$ HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM  BNL-DP     277
360 F4     RS PLXT B$ LEOPARD, SPECTRA CALCULATION WITH DEPLETION    WAPD     279
360 F4     RS PLXT B$ XSDRN, DISCRETE ORDINATE MULTIGROUP CONSTANTS  ORNL     393
360 F4     RSB L T M$ CLIP, FORM OR THREDES LIBRARY UTILITY ROUTINE  AI       271
360 F4     RSBP  T K$ THREDES, 1-D FEW-GP DIFFUSION DESIGN SYSTEM    AI       273
360 F4     RSBP XT D$ SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION     AI        58
360 F4     RSBPLXT C$ ANISN, 1-D MULTI-GP DISCRETE ORDINATE CALC     AI       151
360 F4     RSBPLXT C$ CAESAR4,LIBLST, 1-D MULTI-GP DIFFUSION + LIB   AI       270
360C ASAF4 RS P  T M$ ETOE, ENDF/B TO MC**2 DATA CONVERSION          APDA     350
3600 F36     SBP     I$ BOW2, DEFLECTION CALCULATION PARALLEL BEAMS  ANL      365
3600 F36     RS P  T B$ MC**2, ENDF MULTIGROUP X-SECTION CALCULATION ANL      355
3600 F36     RS P  T C$ SNARG-1D, 1-D MULTI-GP DISCRETE ORDINATE CALC ANL     288
3600 F36     RS P  T I$ SUPORAN, REACTOR CORE SUPPORT STRESS ANALYSIS ANL     357
3600 F36     RS P XT C$ MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE ANL     262
3600 F36     RSBP    A$ MISH-MASH, RESONANCE INTEGRAL CALC HOMOGENEOUS ANL   214
3600 F36     RSBP    G$ CHEMLOC2, CORE HEATING CLADDING-STEAM REACTION ANL   366
3600 F36     RSBP    G$ WHAM, LIQUID-FILLED PIPING SYSTEM ANALYSIS     KE     278
3600 F36     RSBP    I$ SWELL2, FUEL ELEMENT LIFETIME ANALYSIS         ANL    353
3600 F36     RSBP XT B$ RABBLE,WLIB,FLAT, RESONANCE ABSORPTION, CELL   ANL    281
3600 F36     RSBP XT B$ RIFF-RAFF, RESONANCE INTEGRAL CALC 2-REG CELL  ANL    213
3600 F4      RSBP  T D$ 2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION    ANL    325
3600 F63     RS  LXT B$ GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC     ANL     33
3600 F63     RSBP    A$ NEARREX, COMPOUND NUCLEUS X-SECTION CALC       ANL    171
3600 F63     RSBP    E$ R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS   ANL    255
3600 F63     RSBP    E$ R102, SPACE-INDEPENDENT INVERSE KINETICS CALC  ANL    168
3600 F63     RSBP    F$ AX1, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE   CDC    102
3600 F63     RSBP    C$ COINC, COINCIDENCE COUNTING DATA REDUCTION     ANL    248
3600 F63     RSBP  T A$ 2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS  ANL    254
3600 F63     RSBP  T C$ FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION   CDC    167
3600 F63     RSBP  T E$ TRAFICORPORATION, TRANSFER FUNCTION SYNTHESIS  ANL    135
3600 F63     RSBP  T G$ R153, PARAMETRIC SITE REQUIREMENT STUDY        ANL    172
3600 F63     RSBP  T H$ ARGUS, TRANSIENT TEMPERATURE CALC CYLINDER     ANL    152
3600 F63     RSBP  T M$ XLIBIT, X-SECTION LIBRARY UTILITY ROUTINE      ANL    181
3600 F63     RSBP XT B$ THERMOS(ANL), THERMAL SPECTRUM X-SECTION CALC  ANL    184
3600 F63     RSBPL T J$ MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY    ANL    143
4RESTRAINT PIPE STRESS, MAXIMUM MOMENT CALC     AI    7090 F+FAP RS    T I $  109
48 GAFGAR ENDF/B DATA TAPES    GGA  1108 BIN    R   L T Z$ ZPR-III ASSEMBLY   356
500PSIA 32-472DEGF BAPL 6600 F4     RS    XT H$ WATER, STEAM TABLES 14.5-14, R267
625 F4     RS     T C$ EXTERMINATOR2, 2-D MULTI-GP DIFFUSION PROGRAM BC     156
625 F4     RS     T C$ FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE BC     28
625 F4     RS P     B$ AVOID, ANNULAR VOID X-SECTION CALCULATION      GEC    276
625 F4     RS P  T C$ DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM   BC     209
625 F4     RS P XT B$ GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV    BC     185
635 F+FAP RSBP  T Q$ CASCADE,CLUSTER, RADIATION DAMAGE IN METALS     GEC    419
635 F+GMP RSBP  T H$ THTE, 3-D TRANSIENT HEAT TRANSFER PROGRAM       GEC    346
635 F4     RS P     D$ RAPFU, FUEL CYCLE PARAMETERS FAST BREEDERS     APO    372
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635	F4	RS	P	T	B$	REAX, RESOLVED RESCNANCE EPITHERMAL X-SECTIONS	NED	257
635	F4	RS	P	T	C$	FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION	NED	167
635	F4	RS	P	T	H$	MANTA, STEADY-STATE THERMAL-HYDRAULIC ANALYSIS	NED	256
635	F4	RS	P	T	O$	BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS	GEV	311
635	F4	RS	P	T	O$	EXPN, ANALYSIS OF PULSED NEUTRON SOURCE DATA	NED	258
635	F4	RS	P	XT	F$	FORE2, FAST REACTOR EXCURSION CALCULATIONS	NED	174
635	F4	RS	PL	T	C$	BISYN, 2-D MULTI-GP DIFFUSION SYNTHESIS CALC	NED	287
635	F4	RSBP			D$	CHAINS, ANALYSIS OF RADIOACTIVE DECAY CHAINS	GEC	418
635	F4	RSBP			D$	DCS, NEUTRON FLUX-DOSIMETER ACTIVITY RELATION	GEC	423
635	F4	RSBP			O$	MICHRD, MICROHARDNESS MEASUREMENT ANALYSIS	GEC	421
635	F4	RSBP		T	B$	GROUSE, SPACE-DEPENDENT X-SECTION GENERATION	GEC	420
6400	F4	RS	P	T	G$	ECCSA1, LOSS-OF-COOLANT + EMERGENCY COOLING	BCL	330
6400	F4	RS	P	T	G$	NURLOC-1.0, LOSS-OF-COOLANT THERMAL ANALYSIS	BCL	328
6500	F4	RS	P		B$	PARTI, OPTIMAL GROUP OR MESH COLLAPSING	TRW-MMU	416
6500	F4	RS	PLXT		C$	MACH1, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	PURD	262
6600	BCD	R		L T	Z$	ETOG1 DATA LIBRARIES, MUFT4 OR 5 + GAM1 + GAM2	WNES	447
6600	F	R	B	T	M$	TIGIR2, MODULAR DOCUMENT INFORMATION SYSTEM	KAPL	R403
6600	F+ASC	RS			P$	SIMPLE1, TIME-SHARING PROGRAMMING LANGUAGE	KAPL	R442
6600	F+ASC	RS	P	T	H$	LICN, 3-D TEMPERATURE DISTRIBUTION PROGRAM	KAPL	R299
6600	F+ASC	RS	P	T	L$	DATATRAN 2-D GEOMETRY INPUT, PREPARATION, EDIT	KAPL	R406
6600	F+ASC	RS	P	T	P$	CURFIT, CURVE FITTING EXPERIMENTAL DATA POINTS	KAPL	R 43
6600	F+ASC	RS	P	T	P$	DATATRAN UTILITY MODULES, 2-D + 3-D PLOTTING	KAPL	R407
6600	F+ASC	RS	P	T	P$	DOGGY, DESK CALCULATOR FORM SHEET DP PACKAGE	KAPL	R428
6600	F+ASC	RS	P	T	P$	LIZARD, NONLINEAR DIFFERENTIAL EQNS. SOLUTION	KAPL	R445
6600	F+ASC	RS	P	T	P$	SNEQ, NONLINEAR ALGEBRAIC EQN SOLN CURVE PLOT	KAPL	R364
6600	F+ASC	RS	P	XT	D$	PDQ7, 1,2 OR 3-D FEW-GP DIFFUSION DEPLETION	BAPL	R275
6600	F+ASC	RS	P	XT	J$	ASPIS, GAMMA RAY SOURCE BUILDUP FACTOR CALC	BAPL	R429
6600	F+COM	RS	P	XT	C$	FARED, 1-D FAST REACTOR DESIGN & SURVEY STUDY	B&W	427
6600	F+COM	RS	P	XT	I$	GAPL3, INELASTIC LARGE DEFLECTION STRESS STUDY	BAPL	R397
6600	F+COM	RSB		T	M$	DATATRAN, MODULAR PROGRAMMING AND DATA SYSTEM	KAPL	R386
6600	F4	RS			I$	SORSDB, PRESSURE VESSEL STRESS AND FATIGUE	KAPL	R391
6600	F4	RS		T	M$	CHECKER,CRECT,CAMMET,PLOTFB,SLAV3, ENDF/B PROC	NCSC	384
6600	F4	RS		XT	B$	MO807, 2-D DIFFUSION ABSORPTION REMOVAL X-SECS	BAPL	R280
6600	F4	RS		XT	C$	MO648, 1-D SLAB TRANSPORT WITH SLOWING DOWN	BAPL	R342
6600	F4	RS		XT	H$	WATER, STEAM TABLES 14.5-14,500PSIA 32-472DEGF	BAPL	R267
6600	F4	RS		XT	I$	CYGRO2, STRESS ANALYSIS CYL FUEL ELEMENT	BAPL	R266
6600	F4	RS		L T	B$	GGC4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA	GGA	298
6600	F4	RS	P		A$	AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC	BNL	376
6600	F4	RS	P		A$	SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC	BNL	377
6600	F4	RS	P		D$	CINDER,MO102, POINT DEPLETION FISSION PRODUCT	BAPL	313
6600	F4	RS	P		F$	RAUMZEIT, 1-D TIME-DEPENDENT DIFFUSION CALC	KAPL	R352
6600	F4	RS	P		F$	STINT3, SINGLE-CHANNEL SPACE-TIME SYNTHESIS	KAPL	R389
6600	F4	RS	P		H$	RAPP, HIGH-VELOCITY FLOW STUDY STEAM-WATER MIX	KAPL	R382
6600	F4	RS	P		I$	STEM, MATRIX GENERATION FOR A SYSTEM OF BEAMS	KAPL	R337
6600	F4	RS	P		I$	TUBE, U-TUBE HEAT EXCHANGER STRESS ANALYSIS	KAPL	R378
6600	F4	RS	P		N$	HEATMESH, GEOMETRICAL DATA HEAT TRANSFER STUDY	SLL	434
6600	F4	RS	P		N$	LARCA, FLUX-WEIGHTING OF DTF4 CROSS SECTIONS	LASL	409
6600	F4	RS	P		P$	MOST, A MULTIDIMENSIONAL OPTIMIZATION SCHEME	KAPL	R446
6600	F4	RS	P		P$	ROPE, FINDING ROOTS OF A POLYNOMIAL	KAPL	R444
6600	F4	RS	P	T	A$	TOR, THERMAL SCATTERING CRYSTALLINE MATERIALS	LASL	360
6600	F4	RS	P	T	B$	ETOM1, ENDF/B FORMAT TO MUFT FORMAT X-SECTIONS	WNES	436
6600	F4	RS	P	T	E$	GLEN, GROUP CONSTANT CALC FROM TOR OUTPUT DATA	LASL	361
6600	F4	RS	P	T	C$	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	LASL	209
6600	F4	RS	P	T	C$	TWOTRAN, 2-D MULTI-GP TRNSPT CODE XY RZ RTHETA	LASL	358
6600	F4	RS	P	T	C$	VARI-QUIR, TIME-DEP 2-D MULTI-GP DIFFUSION	WANL	212
6600	F4	RS	P	T	D$	2DB, 2-D MULTIGROUP DIFFUSION AND DEPLETION	LASL	325
6600	F4	RS	P	T	F$	NCAH, 1-D ONE-GP SPACE-TIME DIFFUSION FEEDBACK	BNL	405
6600	F4	RS	P	T	F$	SAS1A, FAST REACTOR POWER AND FLOW TRANSIENTS	ANL	400
6600	F4	RS	P	T	I$	GEM, EIGENVALUE PROBLEM FOR VIBRATING SYSTEMS	KAPL	R344
6600	F4	RS	P	T	I$	SOR2, STRESS ANALYSIS SHELLS OF REVOLUTION	KAPL	R 80
6600	F4	RS	P	T	O$	JITER, FLUCTUATION EXPERIMENT ANALYSIS	BAPL	R394
6600	F4	RS	P	X	I$	FINEL, FINITE-ELEMENT STUDY 2,3-D STRUCTURES	KAPL	R404
6600	F4	RS	P	XT	A$	PUN1, UNRESOLVED RESCNANCE INTEGRALS X-SECS	BAPL	R359
6600	F4	RS	P	XT	A$	SUMOR, S-WAVE NEUTRON X-SECTION CALCULATION	BAPL	R399

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6600 F4     RS P XT B$ PAX02, HARMONY-PDQ X-SECTION GENERATION CODE    BAPL  R426
6600 F4     RS P XT B$ RESQ2,RESQO,DBF1, RESONANCE INTEGRAL HEX CELL   BAPL  R285
6600 F4     RS P XT C$ BE21, FEW-GP DISCRETE ORDINATES SLAB GEOMETRY   BAPL  R398
6600 F4     RS P XT F$ NOWIG, 1-D 2-GP KINETICS TEMPERATURE FEEDBACK   BAPL  R371
6600 F4     RS P XT F$ TWIGL, 2-D 2-GP SPACE-TIME DIFFUSION FEEDBACK   BAPL  R338
6600 F4     RS P XT F$ WIGL2, 1-D 2-GP SPACE-TIME DIFFUSION 3-GEOM     BAPL  R274
6600 F4     RS P XT G$ FLASH3, LOSS-OF-COOLANT ACCIDENT ANALYSIS       BAPL  R295
6600 F4     RS P XT G$ FLASH4, FULLY-IMPLICIT TRANSIENT SIMULATION     BAPL  R448
6600 F4     RS P XT G$ MO555,ACT1, LOSS-OF-COOLANT ACCIDENT ANALYSIS   BAPL  R284
6600 F4     RS P XT H$ FIGRO, LSBR FUEL SWELLING TEMPERATURE STUDY     BAPL  R272
6600 F4     RS P XT H$ FLOT1,MO219, PWR FLOW TRANSIENT ANALYSIS        BAPL  R331
6600 F4     RS P XT H$ GLUB1, WATER-LOGGED FUEL ELEMENT ANALYSIS       BAPL  R424
6600 F4     RS P XT H$ HOT2, 2-D TRANSIENT HEAT CONDUCTION PROGRAM     BAPL  R286
6600 F4     RS P XT H$ MO899,HOH, STEAM TABLES 14.5-2538 PSIA          BAPL  R294
6600 F4     RS P XT H$ TOPS, TRANSIENT THERMODYNAMICS OF PRESSURIZERS  BAPL  R348
6600 F4     RS P XT H$ WASP, WATER AND STEAM THERMODYNAMIC PROPERTIES  BAPL  R396
6600 F4     RS P XT I$ CYGRO3, OXIDE FUEL ROD STRESS & DEFORMATION     BAPL  R449
6600 F4     RS P XT I$ MANE1, RECTANGULAR MAGNETIC NETWORK SOLUTION    BAPL  R412
6600 F4     RS P XT I$ MO266, LINEAR ELASTIC STRUCTURAL DYNAMICS       BAPL  R383
6600 F4     RS P XT I$ MO457,PIPE, ELASTIC STRESS OF PIPING SYSTEM     BAPL  R329
6600 F4     RS P XT I$ MO552, DYNAMIC ANALYSIS LINEAR ELASTIC SYSTEMS  BAPL  R283
6600 F4     RS P XT I$ SEALSHELL2, SHELL STRESS ANALYSIS AXISYM LOAD   BAPL  R282
6600 F4     RS P XT J$ MO756,LETO, 1-D SLAB GAMMA-RAY TRANSPORT        BAPL  R343
6600 F4     RS P XT L$ BL47, DRAFTING TOOL TO PLOT PLANE STRUCTURES    BAPL  R373
6600 F4     RS P XT O$ REDUX, REACTOR FLUCTUATION EXPERIMENT ANALYSIS  BAPL  R425
6600 F4     RS P XT P$ DAFT1, LEAST SQUARES FIT FISSILE NUCLIDE DATA   BAPL  R327
6600 F4     RS P XT P$ MO661,MO657,MO626, POLYNOMIAL CURVE FITTING     BAPL  R411
6600 F4     RS PL T B$ ETOG1, ENDF/B TO MUFT, GAM, ANISN X-SEC FORMAT  WNES   437
7030 F4     RS     T H$ RATH, 2- OR 3-D HEAT CONDUCTION LUMPED MASS    LASL   242
7030 F4     RS P   T C$ DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM   LASL   209
7030 F4     RS P   T D$ DTF-BURN, 1-D MULTI-GP DTF4 WITH DEPLETION     LASL   269
7040 F+MAP  RS P   T G$ CONTEMPT, LOSS-OF-COOLANT ACCIDENT ANALYSIS    PPCO   297
7040 F+MAP  RS P   T G$ RSAC, RADIOLOGICAL SAFETY ANALYSIS PROGRAM     PPCO   265
7040 F4     RS P     A$ COMBCO, COMBINED B-W MULTI-LEVEL CONVOLUTION    PPCO   203
7040 F4     RS P   T C$ TOPIC, 1-D FEW-GP SN APPROXIMATION CYLINDER    PPCO   148
7044 F+MAP  RS P   T B$ GAROL, RESONANCE OVERLAP AND LATTICE EFFECTS   GGA    219
7044 F+MAP  RS P   T D$ OPLS, POWER PLANT PERFORMANCE AND PRICE STUDY  GGA    226
7044 F+MAP  RS P   T H$ TOODEE, 2-D TIME-DEPENDENT HEAT CONDUCTION     PPCO   349
7044 F+MAP  RSBP     M$ UNPACK, RETRIEVAL FROM SCISRS X-SECTION TAPE   GGA    206
7044 F+MAP  RSBP   T C$ GAMBLE4, 2-D MULTI-GP DIFFUSION XY RZ GEOMETRY GGA    222
7044 F4     RS       I$ CORE, CORE CONFIGURATION FUEL TEMPERATURE CODE GGA    233
7044 F4     RS     T H$ BLOOST5, POINT-KINETICS WITH 2-D HEAT TRANSFER GGA    205
7044 F4     RS    XT G$ PRECON, HTGR CONTAINMENT PRESSURE POST RUPTURE GGA    228
7044 F4     RS P     A$ FASDOP, X-SECTIONS FROM B-W RESONANCE DATA     GGA    216
7044 F4     RS P     D$ RAD2, HTGR FISSION PRODUCT ACTIVITY DIST STUDY GGA    231
7044 F4     RS P     D$ WAMPUM, FUEL CYCLE COSTS PERFORMANCE STUDY     GGA    224
7044 F4     RS P     I$ GAZELLE5, GAS-COOLED CORE THERMAL DESIGN STUDY GGA    232
7044 F4     RS P     L$ GAVER, ENERGY INTERVAL X-SECTION AVERAGING     GGA    218
7044 F4     RS P   T B$ FMC-N, MONTE CARLO CALC NEUTRON HISTORIES      GGA    195
7044 F4     RS P   T C$ GASP2, 1-D FEW-GP DIFFUSION POWER DIST SEARCH  GGA    220
7044 F4     RS P   T C$ TEMCO, 1-D FEW-GP DIFFUSION TEMP COEF CALC     GGA    225
7044 F4     RS P   T D$ GARGOYLE, FUEL CYCLE ANALYSIS PARTIAL REFUEL   GGA    260
7044 F4     RS P   T D$ RELOAD-FEVER, 1-D FEW-GP DIFFUSION DEPLETION   GGA    221
7044 F4     RS P   T D$ REVISED-GAD, FUEL CYCLE ANALYSIS W/REFUELING   GGA    223
7044 F4     RS P   T D$ STMGEN, STEAM GENERATOR DESIGN CRITERIA COSTS  GGA    227
7044 F4     RS P   T G$ ELBOW, FUEL ELEMENT STRESS ANALYSIS STUDY      GGA    229
7044 F4     RS P   T G$ RELAP2, REACTOR BLOWDOWN - EXCURSION ANALYSIS  INC    369
7044 F4     RS P   T I$ SAFE-AXISYM, STRESS ANALYSIS AXISYMMETRIC LOAD GGA    251
7044 F4     RS P   T I$ SAFE-PORS, STRESS ANALYSIS AXISYMMETRIC LOAD   GGA    250
7044 F4     RS P X   A$ LEGCOEF3, LEGENDRE COEF CALC FOR ANGULAR DIST  GGA    217
7044 F4     RSBP   T N$ CROSSPLOT, SC4020 PLOTS FROM X-SECTION TAPES   GGA    207
709 F2     RS       C$ FIRN, 2-D FEW-GP S4 APPROXIMATION RZ GEOMETRY  LRL      7
7090 F+FAP RS      T I$ 4RESTRAINT PIPE STRESS, MAXIMUM MOMENT CALC   AI     109
7090 F+FAP RS  L T B$ QUICKIE, INFINITE MEDIUM SPECTRUM X-SECTIONS    AI     119
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7090	F+FAP	RS		L	T	C$	FAIM, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	AI	120
7090	F+FAP	RS		L	T	C$	ULCER, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	AI	118
7090	F+FAP	RS	P		T	B$	RAFFLE, 1ST FLIGHT COLLISION PROBABILITIES MC	ORNL	392
7090	F+FAP	RS	P		T	B$	SOPHIST1/2/5, MULTI-GP TRANSFER COEFFICIENTS	LRL	160
7090	F+FAP	RS	P		T	C$	EXTERMINATOR, 2-D MULTI-GP DIFFUSION PROGRAM	ORNL	156
7090	F+FAP	RS	P		T	C$	ASSAULT, 2-D MULTI-GP DIFFUSION DEPLETION CODE	ORNL	240
7090	F+FAP	RS	P		T	M$	DFSR, DATA FILE SERVICE ROUTINES ENDF TAPES	BNL	236
7090	F+FAP	RS	PL		T	B$	GAMTEC2, MULTI-GP CONSTANT CALC 0 TO 10 MEV	BNW	185
7090	F+FAP	RS	PL		T	B$	TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS	AI	50
7090	F+FAP	RS	PL		T	C$	AIM6, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE	AI	29
7090	F+FAP	RSB			T	A$	TUZ, UNRESOLVED REGION RESONANCE INTEGRAL CALC	GGA	42
7090	F+FAP	RSB			T	A$	ZUT, RESOLVED REGION RESONANCE INTEGRAL CALC	GGA	41
7090	F+FAP	RSBP			T	D$	DDB, 2-D FEW-GP DIFFUSION BURNUP RZ GEOMETRY	GGA	99
7090	F+FAP	RSBPL			T	C$	CRAM, 1-D AND 2-D MULTI-GP DIFFUSION PROGRAM	UK-R	103
7090	F+MAP	RS	P		T	C$	TDSN, 2-D MULTIGROUP DISCRETE ORDINATE PROGRAM	LER	312
7090	FAP	RSBP		XT		Q$	SPECTRA, MC CALCULATION IRRADIATED MATERIAL	GEC	422
7090	FLOCO	RS	P			B$	ZCT, GROUP-COLLAPSING OF MULTI-GP X-SECTIONS	LASL	113
7090	FLOCO	RSBP				C$	2DXY, 2-D MULTI-GP SN APPROXIMATION XY GEOM	AGC	18
7090	FLOCO	RSBP			T	C$	GE-HAPO-S13, 1-D MULTI-GP DOUBLE SN APPROX	BNW	75
7090	F2	RS				A$	LYNNE, WOODS-SAXON POTENTIAL SHAPE CALCULATION	ORNL	381
7090	F2	RS				B$	CANCOFF JR, MODERATOR SPACE CHORD DIST FUNCT	AEG	150
7090	F2	RS				B$	S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION	AI	53
7090	F2	RS				C$	PERT, 1-D PERTURBATION FOR AIM AND FOG CODES	AI	30
7090	F2	RS				E$	AIREK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK	AI	121
7090	F2	RS				E$	SNAPKIN5/5A, 1-REGION KINETICS SNAP GEOMETRY	AI	122
7090	F2	RS				I$	CROCK, SPACE POWER PLANT DESIGN OPTIMIZATION	AI	112
7090	F2	RS				I$	FUGUE, STEADY-STATE TEMPERATURE VOID FRACTION	AI	48
7090	F2	RS				I$	SHOCK, SPACE POWER PLANT DESIGN OPTIMIZATION	AI	114
7090	F2	RS				J$	FARSE1A, DOSE RATE FROM SNAP SHIELD LEAKAGE	AI	91
7090	F2	RS				J$	MORTIMER, DOSE RATE CALCULATION SNAP GEOMETRY	AI	142
7090	F2	RS				J$	SCARF2, SCATTER FROM RADIATOR FINS SNAP GEOM	AI	110
7090	F2	RS				J$	SCAR1, SCATTER FROM A RING SNAP GEOMETRY	AI	111
7090	F2	RS				N$	WED, W-DSN OUTPUT TAPE EDIT REACTION RATES	UK-W	133
7090	F2	RS			T	A$	SUMMIT, CRYSTALLINE SCATTERING KERNEL CALC	GGA	56
7090	F2	RS		L	T	D$	SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION	AI	58
7090	F2	RS	P			B$	BAM, S4 CYL CELL CODE AND TEMPEST COMBINATION	AI	108
7090	F2	RS	P			B$	BOUNCE, FLUX DIST IN MULTI-PIN FUEL ELEMENT	AGC	237
7090	F2	RS	P			B$	SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY	AI	52
7090	F2	RS	P			C$	MIST, 1-D FEW-GP SN DOUBLE SN APPROX SLAB GEOM	PPCO	59
7090	F2	RS	P			D$	AIMFIRE, URANIUM FUEL CYCLE COST ANALYSIS	AI	55
7090	F2	RS	P			D$	ISOCRUNCH, REACTION DECAY CHAIN ANALYSIS	ORNL	180
7090	F2	RS	P			F$	CONEC, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE	LRL	129
7090	F2	RS	P			G$	CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD	AI	47
7090	F2	RS	P			I$	WIREX, COMPUTER-PRODUCED WIRING LISTS UHTREX	LASL	315
7090	F2	RS	P			J$	GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY	AI	45
7090	F2	RS	P			J$	GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM	AI	46
7090	F2	RS	P			J$	RATRAP, DOSE RATE CALCULATION SNAP GEOMETRY	AI	141
7090	F2	RS	P			O$	CROC90, ML-1 FLUID FLOW EXPERIMENT ANALYSIS	AGC	154
7090	F2	RS	P			P$	FRANTIC, LEAST SQUARES FIT SUM OF EXPONENTIALS	MIT	324
7090	F2	RS	P		T	B$	AGN-SIGMA, CALC OF MULTI-GP TRANSFER MATRICES	AGC	243
7090	F2	RS	P		T	C$	EQUIPOISE3, 2-D 2-GROUP DIFFUSION SLAB CYL	ORNL	39
7090	F2	RS	P		T	C$	EQUIPOISE3A, 2-D 2-GP DIFFUSION CYLINDER SLAB	ORNL	87
7090	F2	RS	P		T	C$	FOG, 1-D FEW-GP DIFFUSION SLAB CYLINDER SPHERE	AI	28
7090	F2	RS	P		T	C$	TOPIC, 1-D FEW-GP SN APPROXIMATION CYLINDER	PPCO	148
7090	F2	RS	P		T	C$	W-DSN, 1-D MULTI-GP SN APPROX SLAB CYL SPHERE	UK-W	132
7090	F2	RS	P		T	C$	WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY	ORNL	32
7090	F2	RS	P		T	C$	2DGRAND, 2-D FEW-GROUP DIFFUSION SLAB CYLINDER	ORNL	40
7090	F2	RS	P		T	L$	DPC, DATA PREPARATION FOR 2-D DESIGN PROGRAMS	LASL	234
7090	F2	RS	PL		T	A$	ARES2, RESONANCE INTEGRAL X-SECTION CALC	AI	89
7090	F2	RS	PL		T	B$	FORM, FAST NEUTRON SPECTRUM X-SECTION CALC	AI	51
7090	F2	RS	PLXT			B$	GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC	GGA	33
7090	F2	RS	PLXT			B$	HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM	BNL-DP	277
7090	F2	RS	PLXT			C$	DTF2, 1-D MULTI-GP DISCRETE ORDINATE CALC	AI	151
7090	F2	RSB				F$	HATCHET, COUPLED NEUTRONICS-HYDRODYNAMICS CODE	AGC	153

7090	F2	RSB	T	D$	FEVER, 1-D FEW-GP DIFFUSION DEPLETION PROGRAM	GGA	117
7090	F2	RSBP		D$	NPRFCCP, FUEL CYCLE COSTS PERFORMANCE DATA	KE	146
7090	F2	RSBP		C$	NUCY, SOLUTION OF NUCLIDE CHAIN EQUATIONS	ORNL	134
7090	F2	RSBP		J$	LIPRECAN1, MC NEUTRON PENETRATION CALCULATION	DAC	123
7090	F2	RSBP	X	C$	HERESY1, LATTICE PARAMETERS HETEROGENEOUS CALC	FMA	136
7090	F2	RSBPL	T	J$	MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY	BNW	143
7090	F2	RSBPL	XT	B$	AGN-GAM, FAST SPECTRUM MULTI-GP CONSTANT CALC	AGC	204
7090	F4	RSBP		A$	MUFFLE, FISSILE NUCLIDE X-SECTION EVALUATION	ORNL	323
7090	MAD	RSB		O$	BURP1, DETECTOR EFFICIENCY POINT SOURCE	UM	164
7090	MAD	RSB		O$	BURP2, DETECTOR EFFICIENCY DISK SOURCE	UM	165
7090	MAD	RSB		O$	BURP3, DETECTOR EFFICIENCY POINT SOURCE	UM	166
7090	MAD	RSB		O$	BURP4, GAMMA-RAY PHOTOFRACTION SOLID CRYSTAL	UM	169
7090	MAD	RSB		O$	BURP5, GAMMA-RAY PHOTOFRACTION WELL CRYSTAL	UM	170
7094	F+FAP	RS	L T	M$	MOMUS, X-SECTION LIBRARY UTILITY PROGRAM	AI	159
7094	F+FAP	RS P	T	B$	EPITHERMOS, SPECTRUM AND X-SECTION CALCULATION	GEV	201
7094	F+FAP	RS P	T	G$	AISITE2, PARAMETRIC SITE REQUIREMENT STUDY	AI	172
7094	F+FAP	RS PL	T	B$	AILMOE, X-SECTION CALC ELASTIC SCAT RESONANCES	AI	147
7094	F+FAP	RS PL	T	B$	GRAVE, GROUP-AVERAGING X-SECTIONS PARAMETERS	AI	162
7094	F+FAP	RS PL	T	B$	TYCHE3, MONTE CARLO SLOWING-DOWN DENSITY CALC	AI	149
7094	F+FAP	RS PL	T	G$	CURIE,DOSE,THUNDERHEAD, EXTERNAL+INTERNAL DOSE	AI	196
7094	F+MAP	RS P		A$	CHAD, LEGENDRE COEF CALC FOR ANGULAR DIST DATA	AI	215
7094	F+MAP	RS P	T	E$	AIROS, SPACE-INDEPENDENT KINETICS W/FEEDBACK	AI	163
7094	F+MAP	RS PL	XT	A$	TRIX1, RESONANCE INTEGRAL X-SECTION CALC	AI	208
7094	F+MAP	RS PL	XT	B$	LASER, SPECTRUM CALC WITH BURNUP CYL LATTICE	WAPD	249
7094	F+MAP	RSBP	T	G$	SATURATED BLOWDOWN2, BLOWDOWN ANALYSIS LOFT	KE	200
7094	FAP	RS P	T	H$	RATH, 2- OR 3-D HEAT CONDUCTION LUMPED MASS	LASL	242
7094	F2	RS P		A$	EXT, X-SECTIONS FROM B-W RESONANCE PARAMETERS	WANL	238
7094	F2	RS P		F$	WEAK EXPLOSION, COUPLED NEUTRON-HYDRODYNAMICS	APDA	145
7094	F2	RS P		H$	FLOW-MODEL, MULTI-CHANNEL 2-D 2-PHASE FLOW	AI	246
7094	F2	RS P		J$	SHOE, SHIELD WEIGHT OPTIMIZATION DOSE CALC	AI	197
7094	F2	RS P		P$	EXPALS, LEAST SQUARES EXPONENTIAL DECAY CURVES	LRL	321
7094	F2	RS P	T	A$	2PLUS, NON-SPHERICAL OPTICAL MODEL X-SECTIONS	AI	254
7094	F2	RS P	T	G$	TRANS-FUGUE1, TRANSIENT FLOW AND HEAT TRANSFER	AI	268
7094	F2	RS P	T	H$	HEATING2, TRANSIENT STEADY-STATE HEAT TRANSFER	AI	198
7094	F2	RS PL	T	I$	CYCLOPS1, THERMODYNAMIC CYCLE ANALYSIS	AI	244
7094	F2	RS PL	T	M$	ECSIL, EXPERIMENTAL NEUTRON DATA LIBRARY	LRL	351
7094	F2	RSB	T	N$	CPS, SC4020 PLOTS FROM SCISRS X-SECTION TAPES	WANL	239
7094	F2	RSBP		H$	AXFLU, HEAT TRANSFER MOLTEN FUEL TUBE BUNDLES	LASL	182
7094	F2	RSBP		H$	AXTHRM, HEAT TRANSFER SOLID FUEL TUBE BUNDLES	LASL	183
7094	F2	RSBP	T	H$	PTH1, BLOWDOWN PRESSURE TEMPERATURE HISTORY	KE	155
7094	F4	RS		E$	R102, SPACE-INDEPENDENT INVERSE KINETICS CALC	WANL	168
7094	F4	RS	T	P$	LASL LEAST SQUARES, GENERAL CURVE FITTING	LASL	62
7094	F4	RS P		A$	AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC	BNL	376
7094	F4	RS P		A$	SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC	BNL	377
7094	F4	RS P		B$	GAMMA-P, PRODUCTION X-SECTIONS FOR GAMMA-RAYS	GEC	235
7094	F4	RS P		E$	R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS	WANL	255
7094	F4	RS P		F$	MARS, 2-D EXCURSION CALCULATION R-Z GEOMETRY	APDA	293
7094	F4	RS P	T	B$	GAROL, RESONANCE OVERLAP AND LATTICE EFFECTS	LER	219
7094	F4	RS P	T	C$	DTF4, 1-D MULTI-GP DISCRETE ORDINATE PROGRAM	LER	209
7094	F4	RS P	T	F$	TSN, SPATIALLY-DEPENDENT REACTOR KINETICS	AI	309
7094	F4	RS P	XT	C$	VARI-QUIR3, 2-D MULTI-GP DIFFUSION XY RZ RTH	WANL	264
7094	F4	RSBPL	XT	D$	NAP, NEUTRON-INDUCED GAMMA-RAY RADIOACTIVITY	IITR	314

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